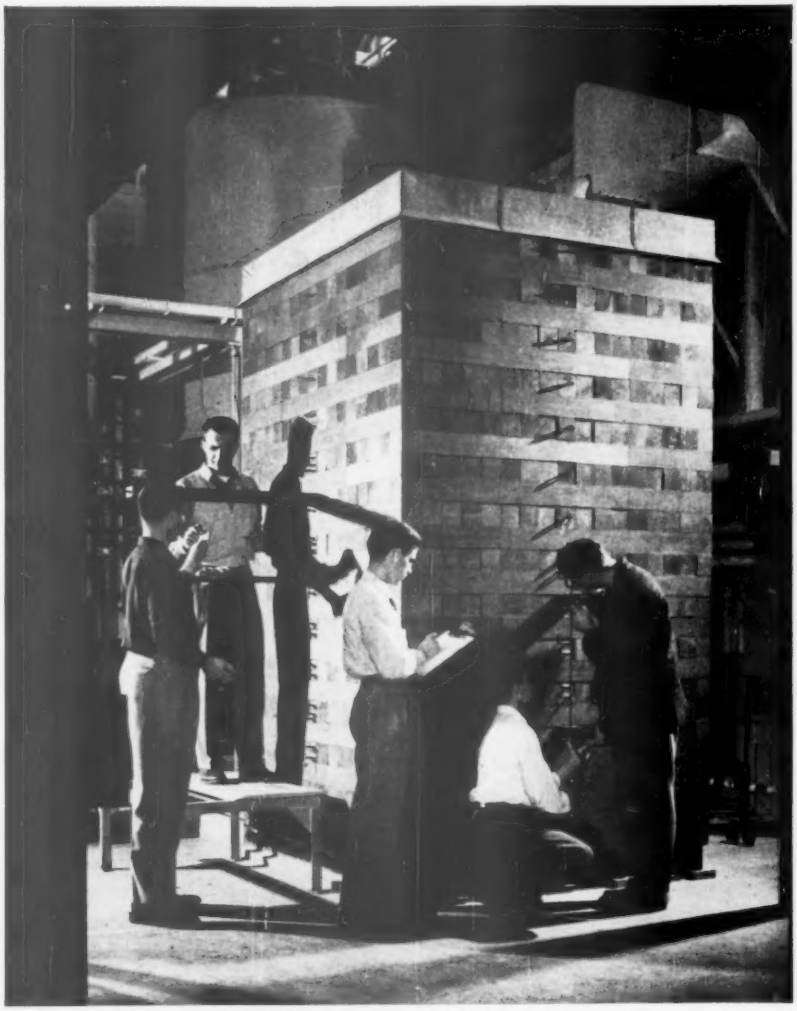
NUCLEONICS

February 1954



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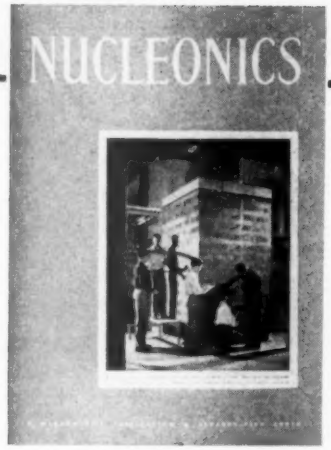

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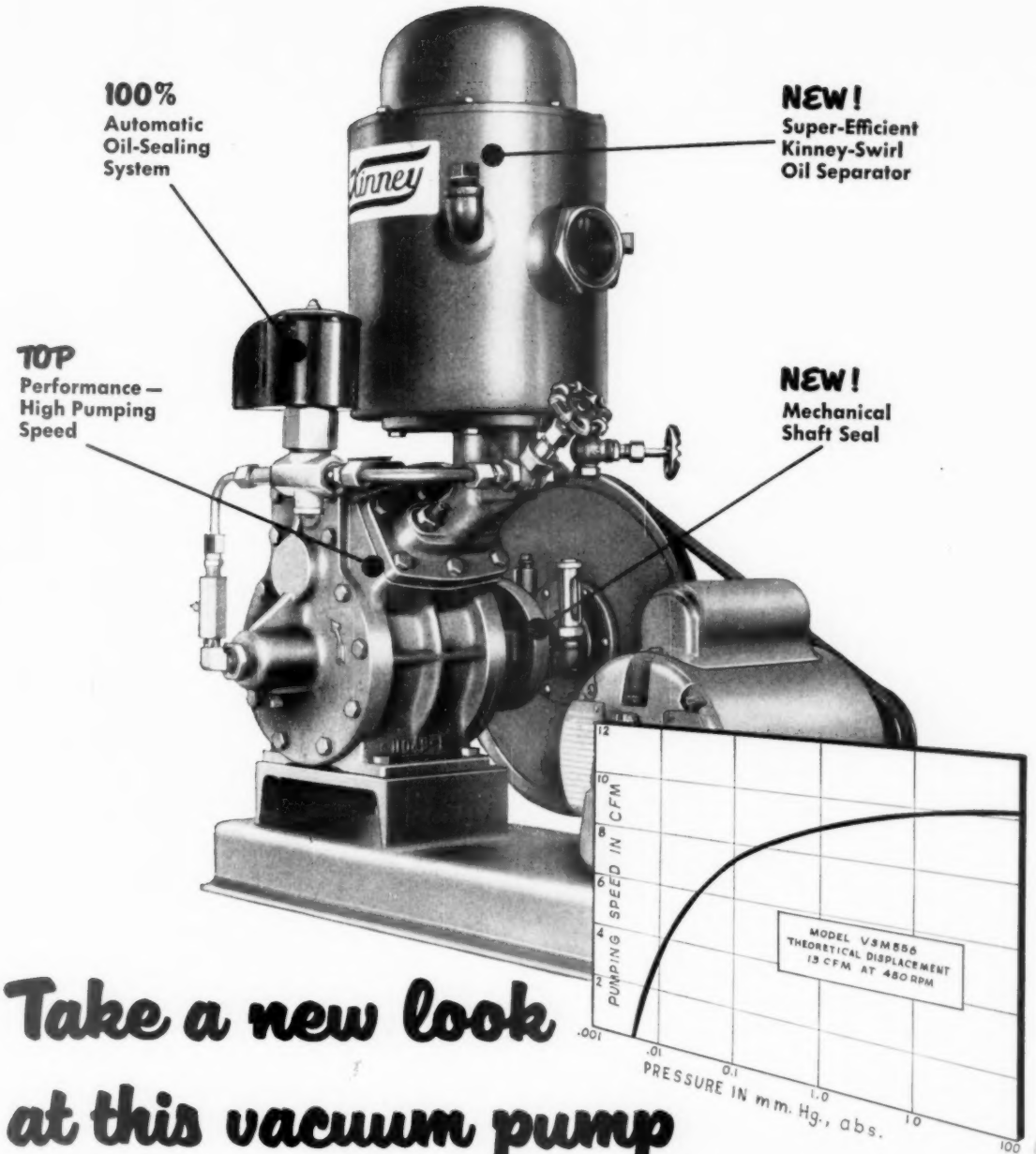

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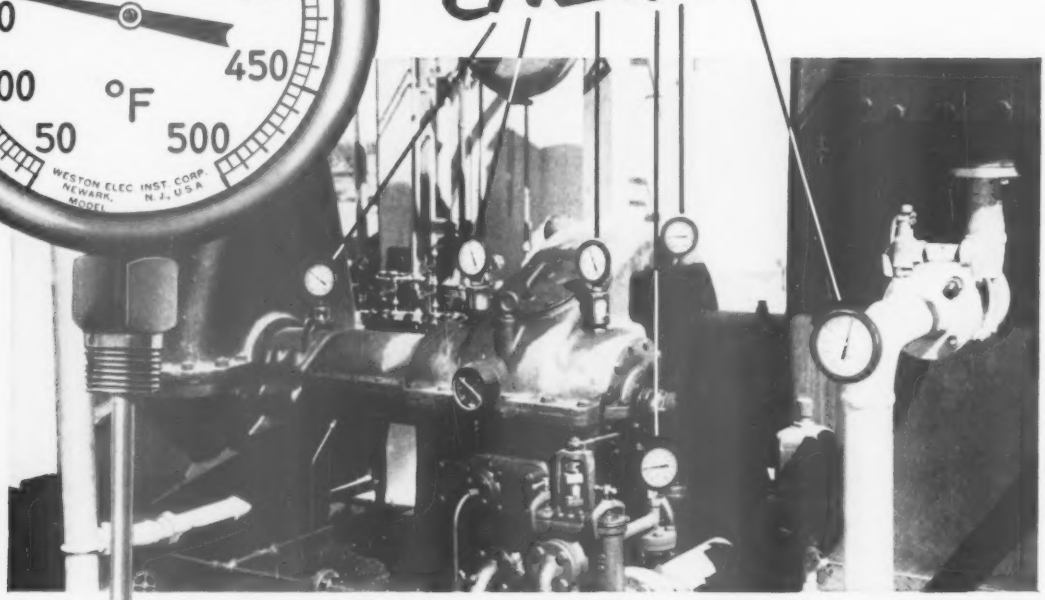

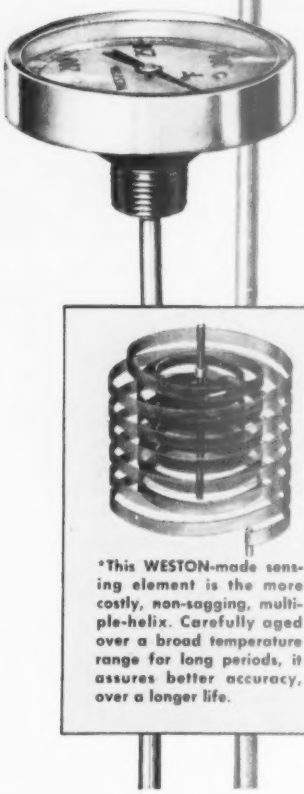

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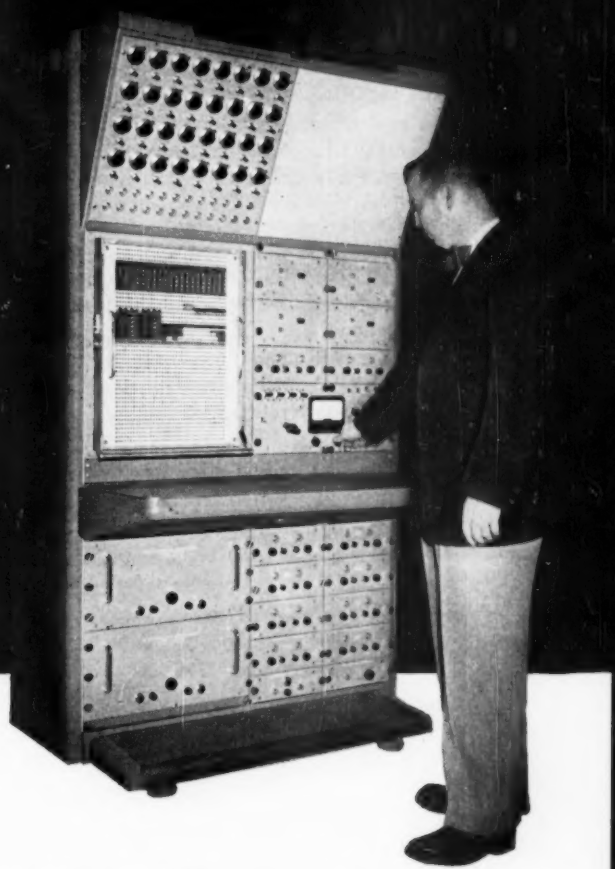

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GENERAL ELECTRIC

NUCLEONICS FEBRUARY, 1954

Jerome D. Luntz, *Editor*

An International Agency

TOOLS, MATERIALS AND KNOWHOW in the atomic energy field are expensive and/or scarce. Nuclear reactors, particle accelerators, uranium —these are the wherewithal of any major nuclear research effort.

Countries that have been immersed in the atomic energy field since World War II days don't suffer any serious lack of these things. Others, handicapped in one way or another, are just barely scratching the surface of this field.

Thus any plan to overcome these limitations would be welcomed by scientists and engineers throughout the world. This is one important light in which to examine the proposal made before the UN by President Eisenhower that an international peaceful atomic energy laboratory be set up.

Substantial precedent already exists for such an agency. First, there is the twelve-nation European Organization for Nuclear Research, which began functioning last year. Its work will be centered around very sizable particle accelerators, too expensive for most of its individual members. Then there is the Joint Norwegian-Dutch Establishment for Nuclear Energy Research whose activities revolve around a nuclear reactor. Here one nation contributed the uranium and the other the heavy water.

BUT NO INTERNATIONAL ORGANIZATION exists on any large scale where both reactors and accelerators are involved. It appears that fissionable material would be an important part of the plan put forth by the President. He said:

> " . . . fissionable material would be allocated to serve the peaceful pursuits of mankind. Experts would be mobilized to apply atomic energy to the needs of agriculture, medicine and other peaceful activities. A special purpose would be to provide abundant electrical energy in the power-starved areas of the world."

Past history on attempts to set up control plans for international use of fissionable material does not augur well for the success of this new plan. But this need not be the case. Previous plans have been directly related to weapons and, of necessity, have called for control of all the world's fissionable material. The Eisenhower plan is different. It does not and need not relate to weapons.

Although we feel it may enhance ultimate chances for atomic weapons control, it can stand on its own feet as a means for pooling world efforts on the peaceful aspects of atomic energy.

Atomic energy was born of international efforts. Many countries around the world are now working hard in the field. Some nations are more handicapped than others. True international efforts on peaceful atomic energy would be a living demonstration around the world that the atom is a symbol of life rather than death.

ONE OF THE PRINCIPAL OBSTACLES to the success of the plan is the secrecy that surrounds data on fissionable material. The U. S. has already taken great strides on relaxing classification of reactor technology. Perhaps the greatest proof of the sincerity of this nation and our collaborators on declassification—England and Canada—would be the releasing of reactor information not related to plants solely useful in the production of fissionable material.

If this were done, the international agency could then operate a laboratory quite similar in scope if not scale to Brookhaven National Laboratory. Brookhaven today is almost entirely an unclassified facility, including much of its 30-megawatt nuclear reactor and, of course, all of its multi-Bev Cosmotron. It is a Mecca for scientists from all over this nation and from many parts of the world—physicists, chemists, biologists, doctors, etc. Much of what Mr. Eisenhower has proposed is already being done at Brookhaven.

In helping power-starved areas of the world, an international laboratory could use its stock of fissionable material to advance the art of reactor technology. Representatives of all member nations would be free to take back with them any of the technology developed in the laboratory, but possibly not any fissionable material. Actually it is this technology that is scarce in some countries rather than source materials, a fact which means that these countries would one day be building power reactors anyway.

In all of the above, we take for granted that the Administration will press implementation of the U. S. Atomic Energy Commission's recommendation that the nuclear power field be open to full industrial participation. This would be further evidence of this country's faith in the peacetime potential of nuclear energy.

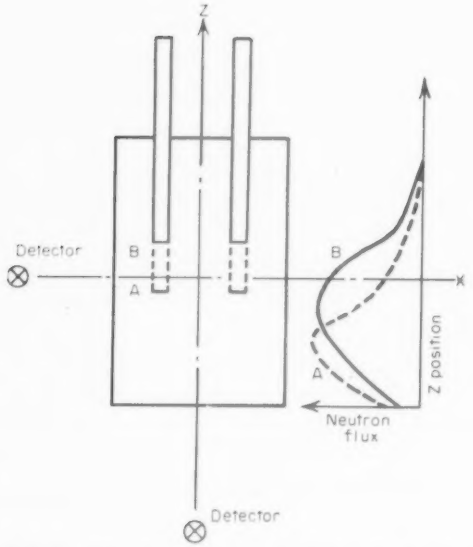

FIG. 1. Side view of four-rod reactor showing how uniform movement of control rods affects vertical flux distribution

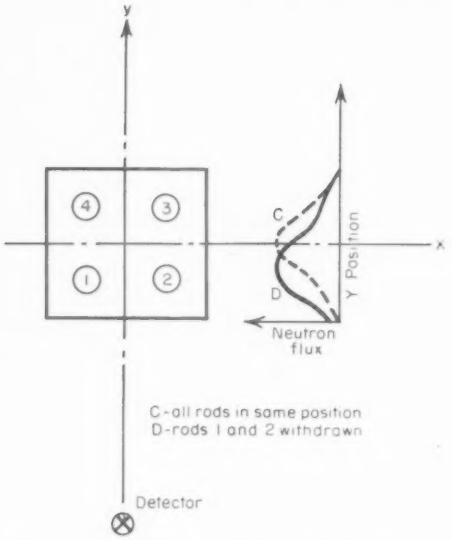

FIG. 2. Top view of four-rod reactor showing how vertical nonuniform movement of control rods affects horizontal flux distribution

Reactor Power Calibration

- Neutron detectors must be calibrated to provide indication of reactor power
- But variables in calibration and operation make continuous absolute output power determination almost impossible
- Thermal measurements look like the best approach to power reactor control

By M. A. SCHULTZ and J. C. CONNOR
Atomic Power Division
Westinghouse Electric Corporation
Pittsburgh, Pennsylvania

REGULATION of reactor power output requires continuous knowledge of the absolute power developed.

Power level of a reactor is generally assumed to be proportional to the number of neutrons in its core. Hence, power level is usually determined by measuring the average neutron flux. However, difficulties discussed in this article limit the usefulness of neutron detectors for continuous power measurement.

Two means are available for continuously measuring average flux. First, a large number of detectors may be placed uniformly throughout the core and an average taken of their readings. The other method involves placing a detector sufficiently far from the core that the core can be considered a point source of neutrons and local neutron density variations within the core will be insignificant.

Problems exist in relating the detectors' output to reactor power. The relation between the power actually developed and the power as reported by neutron detectors or thermal measurements is determined by calibration.

Methods of Calibration

Three basic types of power calibration are available: foil measurements at low power levels, direct energy calibration at low power levels, and conventional calorimetric calibration at high levels. With multiple instrument arrangements, intercalibration of these and neutron detectors is necessary.

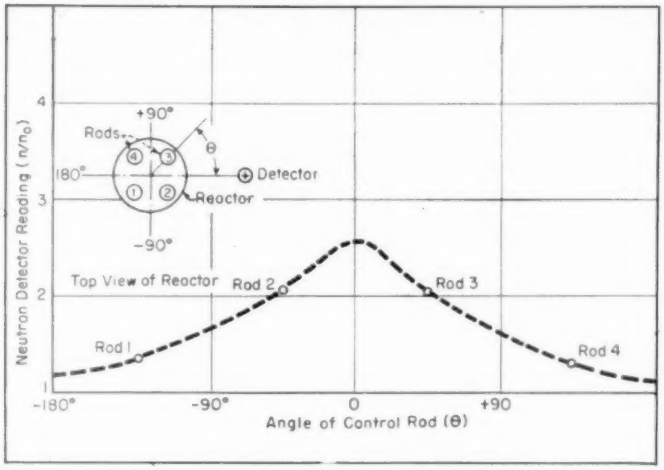

FIG. 3. Subcritical readings vs angle between detector and control rod withdrawn. Counting rate with all rods in $= n_0$; individual rod out $= n$

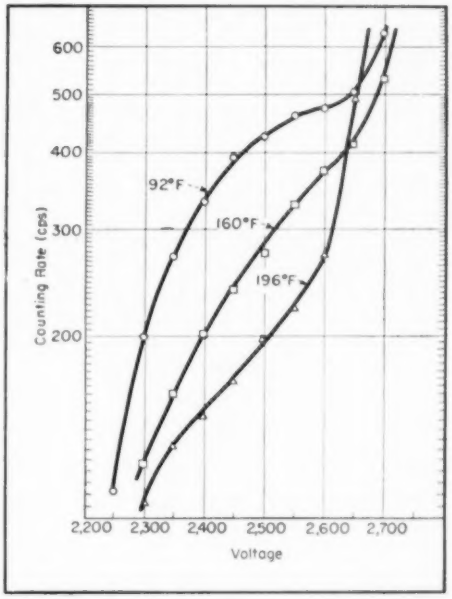

FIG. 4. Effect of temperature on a commercial BF_3 counter

Foil measurements. The first calibration method, experimentally troublesome but capable of yielding reasonable accuracy in a thermal reactor, involves determining average thermal neutron flux by irradiating foils at various locations in the core for a known length of time and measuring the activity induced in the foils (*1*). The power, P, developed within the core is given by the product of the number of fissions occurring per second, and Q, the energy release per fission ($Q = 3.2 \times 10^{-11}$ watt/fission/sec).

$$P = nv\Sigma_f Q \qquad (1)$$

where n is the number of thermal neutrons in the core, v is their velocity in cm/sec, and Σ_f is the macroscopic cross section for fission (calculated for the particular reactor). The usual indium-cadmium foil counting techniques determine only that portion of the neutron flux below an energy of 0.6 ev; however, the average thermal flux can be measured to within 10% by this method. If the core structure is at all complicated, a large number of measurements are required to determine average core flux. Access to the core for removal and insertion of foils must be possible.

Direct energy measurements. A method of low-level calibration that does not require a highly accessible core involves bringing the reactor to exact criticality, inserting a neutron source of known strength, and plotting the resultant rise in counting rate of the neutron instruments as a function of time. By solving the pile kinetic equations (subject to the proper boundary conditions) it can be shown that the rate of change of neutrons with time after an initial transient, and consequently the rate of power increase, is linear and proportional to the source strength, S. In units of n/sec

$$S = K\frac{dn}{dt} \qquad (2)$$

where the constant

$$K = 1 + \frac{1}{l^*}\sum_{i=1}^{6}\frac{\beta_i}{\lambda_i} \cong 1.25 \times 10^4$$

for a graphite-moderated reactor, l^* is the mean neutron lifetime ($\cong 10^{-3}$ sec), β_i is the fraction of fission neutrons emitted by the ith delayed neutron emitter, and λ_i is the decay constant of the ith delayed neutron emitter. Thus

$$\frac{dn}{dt} \cong \frac{S}{1.25 \times 10^4} = 8 \times 10^{-5}S \qquad (3)$$

The expression for dn/dt may be integrated, giving

$$n = n_0 + 8 \times 10^{-5}St \qquad (4)$$

where n_0 is the neutron density at a time $t = 0$ after the initial transient is past and the reactor is rising linearly with time. If T is the time required for the counting rate of a neutron measuring instrument to double

$$n_0 = 8 \times 10^{-5}ST \qquad (5)$$

It can also be shown that $v\Sigma_f$ in Eq. 1 is equal to $1/\epsilon l^*$, where ϵ is the number of neutrons per fission. The final expression for reactor power for the graphite reactor by this method would be

$$P = \frac{nQ}{\epsilon l^*} = \frac{8 \times 10^{-5}ST}{\epsilon l^*} \qquad (6)$$

The accuracy of power calibration measurements made in this manner depend upon the validity of the assumptions made in the derivation, the power level at which the measurements are made, and the strength and calibration accuracy of the neutron source.

Thermal measurements. The difficulties in obtaining a power calibration at low levels largely disappear at high powers where the reactor is developing sufficient heat to allow calorimetric measurements. Intercomparison data is obtained between overlapping low- and high-level neutron instrumentation at some convenient set of power points. The high-power instruments can be calibrated at a given reactor power output by measuring coolant flow and temperature rise across the core. Power is proportional to the product of these two variables. Having obtained a calibration of the high-level instruments, the low-level detectors can be intercalibrated. The accuracy of power calibrations made in this manner may

be as good as that of the temperature and flow measuring instruments.

Flux, Temperature Problems

Having once established the calibration of an instrument, it is necessary to see how this calibration changes with time and operating conditions. For proper calibration of a thermal-neutron detector, the thermal-neutron flux at the detector location must be a function of power only.

Energy variations. As far as the response of a particular instrument is concerned, the reactor may be regarded as a neutron source surrounded by an attenuating medium. The neutron attenuation between the source and a particular point in the shielding structure depends upon distance, the scattering and absorption cross sections of the shielding materials, and the physical arrangement of the shielding structure.

Further, since a reactor is not a monoenergetic neutron source, changes in neutron energy distribution will result in flux variations at the detector even if the neutron emission rate remains constant. The reason for this is that the absorption cross sections of the shielding material are functions of neutron energy.

Distribution variations. Ordinarily, a reactor cannot be considered as a point source of neutrons unless their distribution as a function of position in the core remains constant for all operating conditions. A control rod movement causes a change in neutron flux distribution within the core. It may be necessary or desirable to move control rods to calibrate control rod effectiveness or reactivity, to compensate for fission product poisoning, to compensate for fuel depletion, to compensate for reactor temperature changes, to change flux distribution within the core so that fuel depletion occurs uniformly or in a prescribed manner, or to change the power level. The relative effect of these rod movements on neutron instrument calibration depends upon the design of the reactor.

Figures 1 and 2 show a thermal reactor containing four absorber-type control rods that move parallel to the Z axis. The curves of Fig. 1 show what happens to the neutron flux distribution along the Z axis in the core if all four control rods are moved uniformly from position A to position B. The reading of a neutron detector located beneath the reactor decreases since the peak in flux distribution shifts up and away from the instrument. The reading of a detector located on the X axis would increase since the flux peak moves toward the axis of the instrument.

The curves of Fig. 2 show what happens to the neutron flux distribution along the Y axis for a nonuniform rod withdrawal. Curve C shows the symmetrical flux distribution present if all rods are at a uniform height. Curve D shows how the flux distribution peak shifts toward a detector located on the Y axis if reactivity changes are compensated by withdrawing rods 1 and 2 while 3 and 4 remain at the same height. No detector location exists for a reactor of the design shown that is not subject to some neutron-detector sensitivity shift because of changes in flux distribution.

Instrument calibration changes caused by rod motion are called "rod shadow effects." Planning an experiment to determine the magnitude of a particular effect may be extremely complicated by the simultaneous presence of many other variables, a few of which will be discussed later. Some idea of the order of magnitude of the calibration changes because of rod movements alone may be obtained from Fig. 3. This reactor contains four control rods equally spaced radially from the center of the core. If the reactor contains a separate neutron source, a particular instrument may have a given source level counting rate with the reactor subcritical and all control rods inserted. A plot is shown of the counting rate data obtained by withdrawing and re-inserting one rod at a time as a function of angular position of the rod measured with respect to a line between the detector and the center of the core. The maximum effect occurs for the rods closest to the detector. The dashed curve through the four rod-position points indicates the counting rates that might be obtained if the reactor had more than four control rods.

Rod shadow effects of this nature are greater the closer the instrument is to the core. At points in the shield far from the reactor, the neutron flux is more closely proportional to the total number of neutrons in the core. Unfortunately, choice of distance between core and instrument is usually dictated by instrument sensitivity. However, the highest-power-level detectors may be placed farthest from the core, thus minimizing rod shadow effects for power operation.

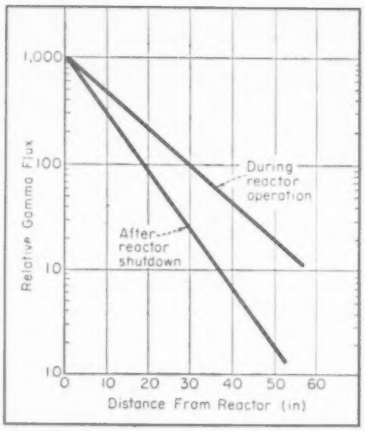

FIG. 5. Relative gamma flux vs distance from reactor. Time is constant. Flux at reactor is taken as 1,000 in each case

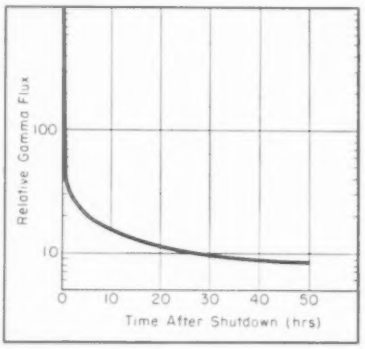

FIG. 6. Gamma decay after shutdown. Curve is for a point at a particular distance from reactor

Temperature variations. Most pile operating conditions, in addition to causing control-rod movement, create even more complex disturbances on the neutron detector calibrations. For example, consider the case of a change in reactor temperature. The neutron flux measured at a detector location in the shielding structure depends to a large extent on the thermalization of fast neutrons from the reactor in the immediate vicinity of the detector. If the temperature of a reactor changes sufficiently to cause the density of the moderator to decrease, the number of fast neutrons reaching the detector will increase. Changes in neutron energy distribution in the reactor may accompany temperature variations. The

leakage flux of fast neutrons, and hence the neutron detector calibrations, depend upon the magnitude of these effects.

Temperature can affect instrument reading directly, even assuming a constant flux. Figure 4 shows the operation of a commercial BF_3 counter at three temperatures. Unless great care is taken in instrument design and usage, counting rate is tremendously dependent on temperature.

Effect of Gamma Rays

An instrument's ability to operate under gamma irradiation is a problem always encountered in determining the usefulness of a neutron-detecting instrument in a reactor system. The existence of large gamma intensities around a power reactor must be taken for granted and provisions made to live with them. The gamma radiation might originate in the decay of fission products, or from reactor materials that capture incident neutrons. The instruments are most likely to be affected by capture gammas in the shield or in the instrument itself. For most purposes, the gamma equilibrium radiation at any instant in the reactor operating cycle can be considered, as a first approximation, to fall off exponentially with distance from the reactor. Gamma-level problems are substantially reduced if the instrument is placed farther away from the reactor. Figure 5 shows a typical gamma flux fall-off with distance for a water-iron shielding mixture under two operating conditions.

Figure 6 shows how the gamma radiation at a particular point in a shield drops off as a function of time after reactor shutdown. The initial sharp drop-off is a function of the decay of fission-product gamma radiation. The slow tail of the curve depends on the half-lives of the materials at the particular point under consideration. The curve is for a structural material containing iron and manganese.

Gamma radiation directly affects the calibration of instruments or circuits in an adverse manner. Several types of neutron-detecting instruments may be used to monitor the neutron level of a power reactor. The effect of gamma radiation is different on each type of instrument.

BF_3 proportional counters. Figure 7 shows a typical integral-bias curve for a BF_3 counter and amplifier operating with and without a superimposed gamma field. With no gamma radiation (other than the normal laboratory background) a conventional plateau-type curve is obtained giving counting rate as a function of discriminator settings. As the gamma background is increased, the tail of the curve at high discriminator settings begins to fall off and ultimately reaches a point where no usable plateau range exists. This point may occur at a gamma background between 50 and 1,000 r/hr, depending upon the design of the counter.

No permanent damage appears to occur to the instrument at these gamma levels. However, at 10^5 r/hr a form of semi-permanent damage occurs. After short exposures to these very high levels the counters do not operate properly and several weeks may be required for the instrument to return to its original operating characteristics. BF_3 counters are ordinarily not used at such high gamma levels. Damage can occur even if the counter is disconnected. The mechanism of this instrument damage is not fully understood, but is related to decomposition of the BF_3 gas and the amount of gas amplification that is obtained in the tube.

Fission counters. No permanent damage occurs in gamma fluxes of 5×10^5 r/hr (*2*). However, the electronic circuits that connect to the fission counter suffer a form of partial blocking because of the immense gamma-pulse pile-up. To minimize the blocking effect it is usually necessary to use extremely short time constants in the electronic circuits. These differentiating networks are generally placed as close to the detecting instruments as possible.

Figure 8 indicates the performance of a typical fission counter under high gamma-radiation conditions. Operation at higher pulse-height settings reduces the gamma contribution to the counting rate much more rapidly than the contribution of the neutrons. Any reduction in neutron sensitivity caused by operating at higher pulse-height settings must be charged directly to the gamma background radiation.

Ionization chambers. A current proportional to neutron flux is usually obtained by the ionization from the alpha particle emitted by the reaction of a thermal neutron with B^{10}. Since the chamber collects all ion pairs created within it, this secondary process must compete directly with the ionization caused by the gamma flux inside the chamber. An ionization chamber can be designed that will have a three- to four-decade neutron range in the presence of a gamma field of 200 r/hr. To extend this range further, it is necessary to compensate the instrument against the gamma radiation. Compensation is usually accomplished by providing two operating chamber volumes. One volume is sensitive to neutrons plus gamma rays, and the other is sensitive to gamma rays alone. A subtraction process enables the instrument neutron range to be extended for roughly two-and-a-half more decades.

Typical response of an ionization chamber operating on a reactor shut-

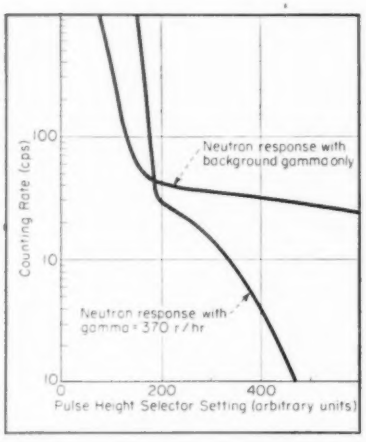

FIG. 7. Responses of typical BF_3 proportional counter with and without gammas

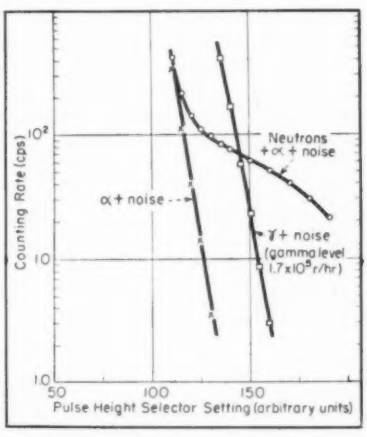

FIG. 8. Response of typical fission counter under three operating conditions

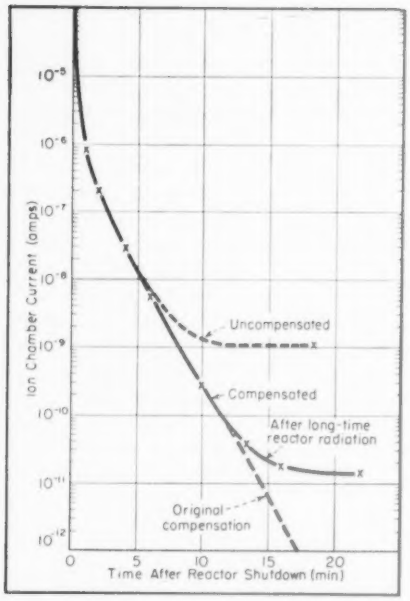

FIG. 9. Response of typical ionization chamber under gamma radiation

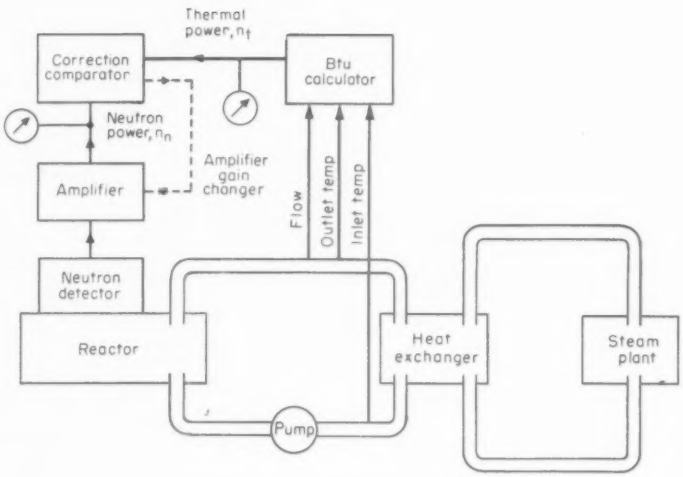

FIG. 10. A continuous power-level calibration system

down problem is shown in Fig. 9. The amount of compensation shifts as a function of time. The shift is caused by changes in the gamma gradient adjacent to the instrument as the reactor builds up the longer-half-life emitters to a fixed activity.

Scintillation counters. It is difficult to obtain a good neutron-to-gamma ratio with these instruments because scintillators are very susceptible to gamma radiation. Because of the temperature-dependence of scintillation counters, their use has been restricted in the thermally-warm vicinity of power reactors.

Thermal Calibration

It has been noted thus far how neutron instrument readings vary against a fixed calibration. A thermal calibration standard itself may vary in the process of intercalibration.

Figure 10 shows a block diagram of a system that attempts to provide a form of continuous power-level calibration of a nuclear power plant. Thermal power in the reactor loop is obtained directly by a "BTU calculator" that computes the product of coolant flow and temperature difference across the reactor. Neutron power is measured by a neutron detector, the signal amplified, and then compared in a correction comparator with the output signal from the BTU calculator. The comparator output adjusts the amplifier gain so that the neutron power reads the same as the calculated thermal power.

The BTU calculator actually reads all the thermal input to the coolant loop. This includes pump power, incidental beta-gamma power from the reactor, and fission power from the reactor. The neutron detector reads essentially fission power as modified by all the local operating conditions previously mentioned. The amplifier-gain changing mechanism sets the neutron power measurement equal to the calculated over-all thermal power measurement at full-power output of the plant. Under these conditions, pump heat input in some plants might be 10% of the calculated thermal output, and beta-gamma power might be of the order of 6% of the total output. The neutron instruments would then automatically be set to read high by a factor of 16% compared with fission power.

Now, a power demand change is made on the system calling for 10% power output instead of 100%. Immediately after this change is made, the beta-gamma power given off by the reactor has not changed appreciably and might represent a full 60% of the actual thermal power output. The pumping losses at low power outputs can probably be considered negligible if the coolant flow is reduced in the power changing operation. The neutron instruments will now be reading fission power of 4% full power plus a 16% correction factor and would indicate that the power output is 4.6% instead of 10%. Ultimately, the BTU-calculator time constants, which should be quite long in order not to disturb the control system, will catch up with this situation and all meters will read 10%. This correction might take a half-hour. Beta-gamma power will eventually decay out and ultimately become 6% of the new power output level, and further continuous corrections would have to be applied to the neutron-measuring instruments.

Because of all the variable factors presented, it is apparent that neutron-measuring instruments are not suitable for power calibrations of nuclear power plants even when fairly complicated correction schemes are used with them. Neutron detectors give only a rough, fast indication of power level and, hence, are more suitable for safety and alarm purposes than for direct power calibration.

* * *

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Why Is Dow-Detroit Edison Working on a Fast Breeder Reactor for Power?

After surveying several types of reactors for industrial power production, this group concentrated on a fast breeder design. The reasons: To make the U. S. independent of uranium imports, and to obtain power economically

By JOHN J. GREBE
Director, Nuclear Research and Development, The Dow Chemical Company, Midland, Michigan

TO MAKE NUCLEAR POWER commercially practical, the Dow-Detroit Edison group started with objectives that seemed impossible to attain. They were even called visionary.

One objective, for example, was to use the available raw uranium far beyond the current yield so as to make this country independent of foreign imports in time of war. At first, reactors that would extend U^{235} by multiplying it a few times were considered good enough, but it was soon realized that this was too shortsighted and narrow a view of present technical capabilities, as compared to those of ten years ago.

The next objective was to make this new source of energy pay its own way by providing an industrial backbone of profitable nuclear energy industry (with new defense capabilities as a by-product) instead of tax-consuming plants. We recognized that if America is to stay afloat in the weapons race, using the old pre-atomic materials and methods on which we are spending most of our defense dollars, all incentive for increasing American productivity will disappear under the overwhelming tax load. It is also inconceivable that our government could develop this new and ultra-potent military potential to its full military utility as long as it is based on present high costs. Industry must cut through the maze of alternatives and arrive at a practical and industrially profitable backlog for defense requirements.

SYSTEMS CONSIDERED

To concentrate our efforts on the most practical design, we considered many types of reactors. We rejected all operating or planned reactor designs as not being feasible by commercial standards. Several promising designs received particular attention.

Slurry Reactor

We attempted to develop a uranium oxide or oxyfluoride slurry reactor, with or without heavy water, using graphite for the core structure. With continuous processing and recycling of the fuel, we thought this unit would breed and might produce power economically. Even though it would be quite workable, the need for expensive low-pressure, low-temperature steam installations caused this reactor to be eliminated.

Turbine conditions would be considerably better with benzene, or the like, instead of steam as the motive fluid. But processing costs with the oxide or oxyfluorides would eliminate any advantages.

Circulating Solid Fuel

Later, we considered a natural uranium unit in which solid metallic fuel would be circulated continuously, using sodium as a lubricant, through pipe coils of zirconium. The moderator would be D_2O. Circulation of the fuel would permit continuous processing. Such a reactor should also breed. The investment in heavy water, however, would be too high to permit this approach, even though other desired conditions could be met.

BNL Liquid-Fuel Reactor

The best thermal-neutron power reactor proposals that we have found are those of Brookhaven National Laboratory. The liquid-fuel processing for

lead-tin-uranium mixtures has been fairly well worked out to permit a minimum processing inventory. The design permits standard steam temperatures and pressures, moderate unit sizes, and low initial costs.

Even if this reactor would not breed, it would not matter too much as long as enriched fuel is available commercially and competitively. The design would have a high conversion ratio in any case. Fuel must be produced by internal conversion of fertile material to fissionable material at a price that is sufficient to pay the whole cost of the enriched fuel. But power cost must not exceed the equivalent of that from 85 coal.

Since it would be very unsound to depend on government-produced fuel, this reactor was not considered good enough to become the basis for industrial power development. However, we count on this general type of reactor system, modified considerably, to produce ten times as much power, in the long run, as is likely to be made from the fast breeder reactors we are designing. It is particularly adaptable to ship propulsion. Breeder reactors should provide the charges for new reactors and the make-up fuel, uranium-233 or plutonium-239, for such thermal reactors for less than the true cost of diffusion plant product.

Some new plans for cost reduction that promise adequate breeding gain make the liquid-fuel thermal reactor important for attaining an immediate objective. The fast breeder should still be followed with diligence for the reasons outlined.

FAST BREEDER

To be reasonably certain of raw material supply in time of war, when power is needed more than ever, we considered it necessary to have a breeder that would operate on residue from current operations rather than to require dependence on foreign uranium resources or government stockpile for make-up fuel.

Three years ago, we concentrated on the fast breeder as the only way to produce power at a low enough cost with the return on capital any new industrial venture would have to pay. We held out for a long time for liquid- or continuously-mobile fuel so as to obtain the lowest possible fuel inventory, continuous reprocessing, and maximum breeding of ultra-quality plutonium. But, because of development problems with containers, radiation stability, processing, etc., we compromised on solid fuel elements early in 1952.

Doubling Time

Fast breeders provide relative freedom in selecting materials of construction. They also permit moderately high temperatures for the power cycle. These advantages alone, however, do not provide enough return for the added fuel invested to compete with thermal reactors having much less uranium in slurries. For these reasons, we regard the minimum doubling time for breaking even to be about three years. This gives a net of about 30% per year on fuel. To provide added incentive, there must be an eventual possibility of more return than this, because it may take many years of gradual development to bring about such yields.

The short doubling time and high breeding gain indicated in these statements may need explanation. Of the three neutrons produced by fission of plutonium, we expect to allow more to be lost than has been usual. This is reasonable because we count on neutrons becoming "cheaper" and we intend to allow greater losses so as to use more simple and economical structural designs. On the other hand, we do not count the neutrons absorbed in uranium-238 or plutonium as a complete loss. Some of the isotopes formed will eventually lead to fission in the reactor proper. It matters very little whether the neutron absorption is high or low for any one isotope in the particular neutron energy range we are dealing with. On the average, the yield of neutrons when the particular isotope produced does fission is great enough to repay partly for the added neutrons previously absorbed. The high neutron absorption is, therefore, only a relatively small disadvantage. The high specific power makes the time at which an equilibrium concentration of the higher isotopes is reached short enough to be tolerated. The self-absorption does, however, provide spontaneous fission to ease the problems of start-up.

To provide extra-quality bomb material, we provide for the added complication of frequent processing of the fuel. The inner and outer blankets must be processed separately, using neptunium from any part of the reactor and only that plutonium produced in the inner blanket for extra-quality bomb material. The plutonium produced in the outer blanket with slower neutrons can be used for replacement fuel, to make up for the fuel that cannot be bred in the reactor core itself.

Inventory and Processing

We want to reduce the inventory of fuel external to the reactor to a few per cent. This is not to infer that all fuel elements will be of exactly the same composition. We may allow for more dilution in the center than at the edge so as to work all parts of the reactor at the same maximum heat-transfer capability. We propose to allow variations in fuel thickness and coolant flow to even out the heat produced and the discharge temperature of the sodium.

The importance of low-cost rapid processing cannot be over-stressed. It should be possible to recover rapidly all the bomb-grade plutonium as neptunium so that it has minimum opportunity of picking up neutrons. The fission byproducts must be recovered while there is still valuable heat and radiation to be obtained from them.

There are still many details to be developed on this method of processing, but ordinary techniques used throughout industry for refining metals also work in this particular field. It is important, however, to recognize that

BACKGROUND

Since January, 1951, Dow Chemical Company and Detroit Edison Company have been cooperating with AEC in studying the feasibility of nuclear power (NU, Feb. '51, p. 86). The group was one of the first four to finance such studies (NU, Jun. '51, p. 42). This particular team has since grown to include 25 companies (NU, Nov. '52, p. 108; June '53, p. 86). They have thus far spent upwards of $1-million of their own money on the project. It is expected that in 1954 they will spend an additional $2-million.

we do not want nor expect thorough decontamination; 90% is more than good enough for our purpose.

We do not expect to purify the plutonium beyond what may be needed for fuel uses. There are ample facilities in operation for chemical purification of the concentrated product.

Optimum alloy composition and exact details of processing will require many years of development. Almost anything we do to improve the strength and corrosion resistance of the fuel will add complication and will also modify the processing requirements.

Fission-Product Uses

To derive a financial return from fission products, we are hoping to develop applications for them as sources of high-temperature heat capable of competing with electro-thermal techniques. Other applications are also expected to be developed.

Safety

The entire apparatus must be designed in such a way that nothing is likely to cause serious trouble. We should be safer around one of these power plants than anywhere else in the city.

We often mention locating the reactor at the foot of Main Street, under a lake or river surface. If cooling water flows through the plant in its natural course and if the plant is protected against bombardment by water serving as self-healing armor, we can't be in bad shape. If, in addition, there are plenty of compressible members consisting of material similar to inner tubes filled with air to protect the structure against shock waves, there isn't much to fear from the outside. Similarly, if the inside of the reactor is designed so that any shock wave approaching it from any direction will cause the fuel to disperse rather than to concentrate, there will be very little to worry about.

To decrease the hazard further, we will put all the heat capacity required for stabilization into the primary sodium cooling circuit. This heat storage and spare cooling capacity can be segregated into small parcels without excessive cost. The heat capacity elsewhere in the system will be maintained at the lowest possible level so that no high-pressure motive fluid can become a hazard because there is so little of it on hand at any time. In fact, our steam boilers, if we use steam, will have an inventory much less than the usual piping within the turbine room.

If benzene or one of the nontoxic and nonflammable fluorinated materials that we are now testing should prove satisfactory, we will be able to eliminate still more complications.

The fuel elements, being designed to operate at a fairly uniform inner temperature, will expand (as a result of phase transformation) in line with the flow of sodium. Therefore, control should be inherent in the design. Even if some of the fuel should melt, no harm would be done because we do not count on keeping the primary cooling circuit free of contamination. In fact, if the entire reactor were to melt down and circulate with the sodium, the design would be such that no dangerous situation would arise. We are even hoping to use this technique as a means of removing fuel elements. By merely shutting down the flow of sodium coolant to any one element, it will melt down for reprocessing.

Cost Data

Power plants never before have had a double-boiler with a temperature-limited source of heat and nice dense media for their motive fluids, so power engineers really don't quite know what to do with this new freedom of choice in materials and design. We expect that this confusion will be cleared up long before we will need to make a decision as to how much we can expect to reduce the cost of the power facilities as a result of this new freedom of choice.

In this instance, efficiency matters very little. Capital cost is by far the most important single factor.

The most important economy may result from the fact that these new heat-transfer fluids are much less corrosive on the lowest cost materials of construction at high and low temperatures than steam. This would permit use of thin-wall small-diameter steel tubing.

Economic evaluation must be based on realistic costs and write-off periods in a new and very rapidly changing art.

It is a long time since power plants improved at the rate of 10% or more per year. Large expenditures in engineering and development are in order when such improvements are possible.

No one can think of current central station plants becoming obsolete in less than 20 years. For this reason, we have had the problem of explaining to our men that the meager cost figures available on government operations really mean very little. We have to make our own estimates of what costs would be if operation were under competitive conditions, where they must face rapid obsolescence, higher cost due to taxes on profits, and, most of all, the need of earning enough net after taxes for the business to grow out of profits.

An industry that must sell itself to the public or bankers to grow has lost its freedom to build for a goal that may take several years of trouble-shooting to attain. Only if the new industry can stand on its own feet and show the capability to expand out of its profits can it get the kind of freedom that is required to do well where chances must be taken. It has been a problem to keep optimists from trying to justify very crude approaches that depend on getting by with the low capital charge of an old, stable and relatively stereotyped industry and apply such charges to something that is bound to change rapidly. We have tried to avoid this trap by looking for the best that could be done under the new circumstances and doing research on what it takes to meet such specifications.

It is easier to continue to improve designs and reduce processing requirements to the point of providing incentives to normal venture capital than to expect sound economics to be set aside by the magic word "atomic." If current tests turn out as favorably as expected, we can expect to interest such venture capital.

It is foolish to think that sound economics can be bypassed with government money. To the best of our knowledge, no new industrial potential has ever been developed by any government, not withstanding enormous expenditures aiming to do so.

Whether or not we will be able to put the best approaches together at one time does not matter. At least, we will know better than to put any more investment into any one phase of this development than is necessary for the purpose of evaluating its capabilities. Each development can be added to the preceding until the whole is assembled over a period of several years, while the reactor, heat disposal, and heat interchange system prove themselves. After satisfactory operation for a year or so, turbo-generators and electric interconnection can be added. END

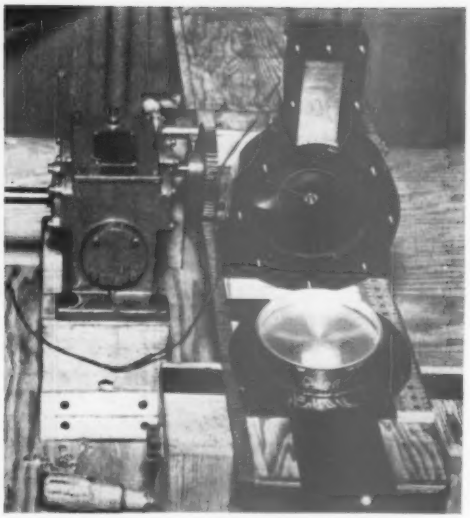

FIG. 1. PV7 photovoltaic cell and plastic scintillator assembly as used in experiments

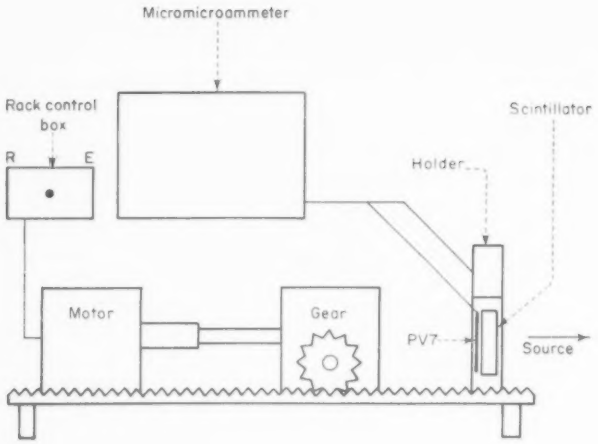

FIG. 2. Diagram of detector system. Rack and pinion used to adjust distance from source to detector

Photovoltaic Scintillation

Linear response from 12,000 to 360,000 r/hr and an estimated accuracy of 4% is achieved with a combination of a photovoltaic cell and liquid or solid scintillators. Simplicity of use and high light output make this unit a practical detector. This article is divided into two sections, the first describing the instrumentation, the second presenting response data for liquid and solid (plastic) scintillator solutions at high dose rates

1. The Detector

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THE DETECTOR ASSEMBLY consists of a General Electric type PV7 photovoltaic cell (4-in. diameter, $\frac{1}{16}$-in. thickness) and a scintillator contained in an aluminized glass cell. These components are mounted in a light-tight Bakelite holder, as shown in Figs. 1 and 2.

Stimulated by incident radiation, the scintillator releases quanta of luminous energy that fall upon the photovoltaic cell, producing a current that is fed directly into a Leeds and Northrup model 9836A d-c amplifier.

The response of this system, exclusive of scintillator, was linear to a PV7 output of 200 μa (the largest current measured) and was unaffected by location of the cell in a high radiation flux. The PV7 has a rated output of 130 μa/mwatt at 4,100 Å. Measurements of photovoltaic-cell background were made over the full dose range and reached 5.5 μa at 360,000 r/hr.

With a scintillator, the maximum PV7 current reached was 151 μa at a sensitivity of 4.2×10^{-4} μa/r/hr.

Scintillator container cells. The cylindrical cells shown in Fig. 3 are designed to provide maximum light collection, with little or no absorption in the end plates. The barrel portion of the cells is cut from Pyrex tubing 92 mm in diameter with a 3-mm wall thickness. The end surfaces are ground carefully to maintain the barrel length at 19 mm and to provide a flat surface for sealing the end plates. To insure excellent optical qualities, the end plates are cut from Kodak type 103-0 spectroscopic plates, 1.27 mm in thickness, with the emulsion removed.

Assembly of the cells requires the following steps since a reflecting coating is evaporated on all internal surfaces except the light-transmitting end plate facing the sensitive PV7 surface.

1. The back plate (farthest from the PV7) is cemented to the barrel with Armstrong A-1 adhesive, and the cement allowed to cure.

2. The aluminum-silicon monoxide coating is evaporated on the internal surfaces.

3. The transparent front plate is sealed to the barrel and the cement allowed to cure.

In actual operation, the plastic scintillator is mounted without a transparent front plate, and the bare plastic

FIG. 3. Cylindrical glass cell to hold scintillator. Ground-glass 14/20 joint filter tube used on liquid cells

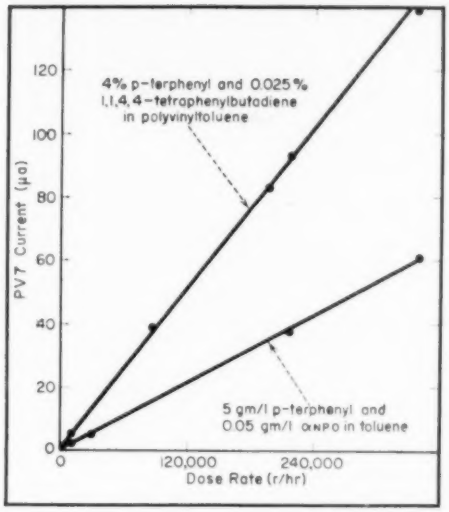

FIG. 4. Light output of plastic and liquid scintillators as function of dose rate shows linear response

Detector for High Dose Rates

rests against the PV7 cell. No noticeable light loss due to absorption is observed when a front plate is used with the plastic.

By simply removing the front cover of the Bakelite holder, one can use both liquid and plastic scintillators.

Instrumentation. The currents generated by the PV7 are measured with a d-c micromicroampere amplifier that has full-scale current sensitivities ranging from 1×10^{-9} to 2×10^{-6} ampere. A shunting arrangement is provided to decrease current sensitivity to the range desired and to keep a constant, low value of external resistance for the PV7 cell. This shunting makes the full-scale current sensitivities range from 1×10^{-7} to 2×10^{-4} ampere with a constant error of 1%. A low fixed total external resistance of 99 ohms insures that the cell current is a linear function of light intensity over the range of intensities encountered.

The PV7 cell was checked for possible sensitivity changes from the high radiation intensities in which the unit is used by replacing the scintillator with a constant light source and observing the output current under all radiation levels. No effect was seen other than the background current measured (maximum of 5.5 μa at 360,000 r/hr) with the PV7 cell alone. Furthermore, the cell showed no fatigue effects when exposed to a constant light source.

2. Scintillator High-Level Response

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SOLUTION SCINTILLATORS of the liquid and solid (plastic) types have been investigated in the very high energy range of 12,000 to 360,000 r/hr (10^6 to 3×10^7 ergs/gm/hr). A radiation source with an estimated activity of 20,000 curies and a gamma spectrum with a mean energy of about 1.3 Mev was used (similar to Co^{60}). Measurements of radiant energy absorption at various distances from the source were made with aqueous ferrous sulfate dosimeters (*1*).

Luminous energy release of the scintillators was measured with the photovoltaic-cell detector.

The plastics and liquids used are listed in the table on page 18. Measured current and thereby the light output of the scintillators appeared to be linear with dose rate over the range 12,000 to 360,000 r/hr. See Fig. 4.

Results are expressed in the table in terms of microamperes per roentgen per hour and also as relative currents. Relative pulse-height data obtained with Cs^{137} internal conversion electrons and a RCA 5819 are included.

It is surprising to see such a marked divergence for the seven cases studied from the low-intensity relative pulse-height measurements to the high-intensity PV7 studies. A calculation for energy response, considering 5,000 photons/Mev are emitted (*2*) independent of radiation rate, fits the better plastic result very well (Appendix). The presence of a slow component in the plastic scintillation pulses could account for their higher current yields.

Conclusions. Two conclusions are:

1. The plastic-PV7 combination will

make a fine dosimeter, at least in the 12,000 to 360,000 r/hr range.

2. The 1 to 12,000 r/hr range must be studied for these scintillators.

The function of the wavelength shifter is demonstrated in the current data by comparing solutions of 5 gm/l *p*-terphenyl in toluene. With 0.05 gm/l αNPO,* the solution gives photons mostly of 4,100 Å; without it, photons of 3,600 Å. This difference works in two manners with the higher wavelength photons: first, to have a greater per cent transmission for the light through the scintillator; second, to have a 30% better current response at the PV7 surface.

The solution of 0.01 gm/l *p*-terphenyl and 0.05 gm/l αNPO in toluene (a good solvent but a shortage of solute molecules) did not lose ground to the plastic scintillators. The solution of 15 gm/l 2,5-diphenyloxazole and 0.05 gm/l αNPO in mineral oil (a poor solvent but an optimum concentration of solute) lost practically all of its ability to scintillate. These facts contribute to the store of information accumulating that will some day give a clear and precise picture of the scintillation process in the liquid phase.

A previously reported detector of this kind using sodium iodide as the scintillator (*3*) was shown to be completely nonlinear in the range of radiation intensity studied.

Appendix

1. Conversion of cell sensitivity from μa/mwatt to μa/photon/sec for 4,100 Å photons:

$$1 \text{ photon} = \frac{6.624 \times 10^{-27} \times 3 \times 10^{10}}{4.1 \times 10^{-5}} = 4.85 \times 10^{-12} \text{ erg}$$

$$1 \text{ photon/sec} = 4.85 \times 10^{-12}/10^{3} = 4.85 \times 10^{-16} \text{ mwatt}$$

$$130 \ \mu\text{a/mwatt} = 130 \times 4.85 \times 10^{-16} = 6.3 \times 10^{-14} \ \mu\text{a/photon sec}$$

2. Conversion of 1 r/hr to the number of photons/sec reaching the PV7: By definition, 83.8 ergs/gm = 1 r; plastic scintillator weight = 100 gm; 1 Mev = 1.59×10^{-6} erg; 1 Mev yields 5×10^{3} photons in the scintillator; and light collection is 90% effective. Thus

$$1 \text{ r/hr} = \frac{83.8}{3600} \times 100 \times \frac{1}{1.59 \times 10^{-6}} \times 5 \times 10^{3} \times 0.9 = 6.6 \times 10^{9} \text{ photons/sec}$$

3. Current resulting from plastic—PV7 combination in a 1-r/hr field:

$$1 \text{ r/hr} = 6.6 \times 10^{9} \times 6.3 \times 10^{-14} = 4.2 \times 10^{-4} \ \mu\text{a}$$

* * *

The authors wish to thank Warren Buck of Argonne National Laboratory for the polyvinyltoluene plastic and Earl Fullman of the Los Alamos Scientific Laboratory for the polystyrene plastic. This article is based on work done under the auspices of the U. S. Atomic Energy Commission.

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* NPO: 2-(1-naphthyl)-5-phenyloxazole.

Comparison of Scintillators at High and Low Intensities

Solution	*High intensity* (μa/r/hr × 10^{-4})	*High intensity* *Relative current*	*Low intensity* *relative pulse height*
PLASTICS			
4% *p*-terphenyl, 0.025% 1,1,4,4-tetraphenylbutadiene in polyvinyltoluene	4.2	1.00	1.00
0.5% *p*-terphenyl, 0.001% αNPO* in polystyrene	2.9	0.69	0.68
LIQUIDS			
5 gm/l *p*-terphenyl, 0.05 gm/l αNPO in toluene	1.8	0.43	0.91
5 gm/l 2,5-diphenyloxazole, 0.05 gm/l αNPO in toluene	1.5	0.36	0.89
5 gm/l *p*-terphenyl in toluene	1.1	0.26	0.82
0.01 gm/l *p*-terphenyl, 0.05 gm/l αNPO in toluene	0.4	0.10	0.09
15 gm/l 2,5-diphenyloxazole, 0.05 gm/l αNPO in mineral oil	0.01	0.002	0.18

Discrepancies between cavity-chamber and free-air measurements of Ra output are resolved by correcting for the chamber-wall density effect. The calculations presented for graphite can be applied to other wall materials

Density

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DETERMINATIONS OF GAMMA-RAY DOSE and of the absorption of γ-ray energy at a point in a solid or liquid medium are generally based on measurements of the ionization in the air volume of a cavity chamber. Bragg (*1*) and Gray (*2*) have shown how the rate of dissipation of γ-ray energy per unit volume of a condensed medium can be related to the ionization in a small air-filled cavity in the medium, taking into account the difference between the atomic numbers of the medium and the air. Gray also showed how these considerations could be applied to the measurement of γ-ray dose in roentgens by means of a small air-filled ionization chamber (cavity chamber) with walls composed of elements of low atomic number.

Gray (*3*) has pointed out that in all of this work it was tacitly assumed that the electron-stopping power of a material depended only on the atomic numbers of its constituents and not on its state of condensation. However, others (*4–7*) have shown that the stopping power for charged particles is less when atoms are closely packed, as they are in a solid, than when they are widely dis-

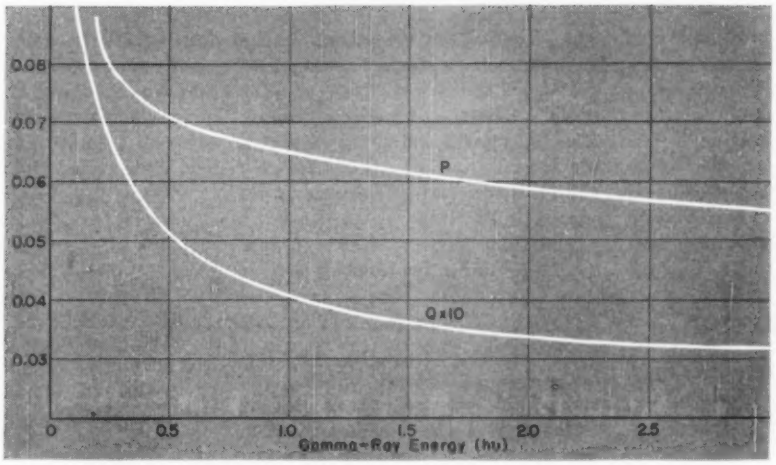

FIG. 1. Variation of P and Q (Eq. 2) with γ-ray energy

Effect in γ-Ray Measurements

persed, as they are in a gas (as a consequence of the reduction of the electric field of the moving particle caused by the polarization of the medium). As a result of this *density effect*, the ionization in a small gas-filled cavity in a condensed medium traversed by γ-radiation is greater than it would be in an identical volume in a gaseous medium of the same atomic number traversed by the same radiation, and an estimate of γ-ray dose based on a measure of this ionization is too high.

What follows is an attempt to calculate the correction that must be applied to cavity-chamber measurements of γ-radiation because of the density effect of the wall material.

Method

The quantity to be determined is the ratio, B, of the ionization produced in an air-filled cavity in a solid material of atomic number Z_w to the ionization that would be produced in an identical air volume surrounded by air at normal temperature and pressure.

Laurence (*8*) has worked out an expression for B based on the Bethe-Bloch formula. To take advantage of his results, it is convenient to divide the calculation into two parts:

1. The derivation of a ratio B', identical with Laurence's expression for B except that the values used for the mean ionization potentials, I, of the wall material and the air are such as to take into account the "zero-energy density effect."*

2. The calculation of a factor D, due to the "finite-energy density effect."*

The first factor depends strongly on the atomic number of the wall material but only slightly on γ-ray energy; the second quantity is relatively insensitive to wall composition but varies markedly with energy.

Low-energy correction. Laurence obtained his expression for B as

$$B = 1 + P(2 \ln Z_w/Z_a) + Q(2 \ln Z_w/Z_a)^2 \quad . \quad . \quad . \quad (1)$$

where Z_w and Z_a are the atomic numbers of the wall material and of air, and P and Q are slowly varying functions of the γ-ray energy, involving the initial Compton-electron energy distribution and the dependence of stopping power on electron energy. The quantities P and Q, calculated from Fig. 2 of Laurence's paper, are presented in Fig. 1.

The factor Z_w/Z_a in Eq. 1 arises from the use of the approximate relationship $I = 13.5Z$ ev for the mean ionization potential I appearing in the stopping-power formula. It is possible to include the zero-energy density effect by using, in place of $13.5Z$, values of I_w determined from experimental stopping powers *in solids* and the value of I_a obtained from air measurements. Then

$$B' = 1 + P(2 \ln I_w/I_a) + Q(2 \ln I_w/I_a)^2 \quad . \quad . \quad . \quad (2)$$

where I_w and I_a are the appropriate mean ionization potentials of the wall material and of air, and B' is written in place of B to indicate that this expression does not take into account the finite-energy density effect. For I_w, the values obtained by Bakker and Segrè (*9*) for 340-Mev protons in solids are suitable; for I_a, one can use the value of 80.5 ev deduced by Livingston and Bethe (*10*) from alpha-particle ranges in air, which agrees well with the recent measurements of Bogaardt and Koudijs (*11*). The quantities $(2 \ln I_w/I_a)$ and $(2 \ln I_w/I_a)^2$ for a number of common wall materials are listed in Table 1. The differences between these values and those obtained by Laurence are not necessarily due entirely to the low-energy density effect,

* The term *zero-energy density effect* refers to the reduction in the electron-stopping power of a condensed material, which is present even for particles of very low energy. The additional effect that sets in at higher energies and increases sharply with energy (*7*) is referred to as the *finite-energy density effect*.

since the assumption that I is proportional to Z probably does not hold exactly, even in the absence of this effect. However, I should take account of any zero-energy effect.

Finite-energy correction. The available calculations on the finite-energy density effect give the reduction in stopping power due to this effect in various materials as a function of electron energy. To estimate the net effect on the ionization in a cavity chamber irradiated by γ-rays, it is necessary to take account of the variation of this reduction with the energy of an electron as it is slowed down in the wall material, the distribution of initial electron energies resulting from Compton interactions of monoenergetic γ-rays, and, finally, the energy spectrum of the actual γ-radiation with which one is concerned. This will be done by a method similar to that used by Laurence in deriving Eq. 1.

Let the probability per incident quantum/cm² of imparting to an electron of the wall an energy between E_0 and $E_0 + dE_0$ at an azimuthal angle between φ and $\varphi + d\varphi$ be $f(E_0)dE_0d\varphi$. Then dn, the number of electrons produced in the volume element $dx\,dy\,dz$ of the material in this interval of energy and angle per quantum/cm², is

$$dn = N_w f(E_0)\,dE_0\,d\varphi \cdot dx\,dy\,dz \quad (3)$$

where N_w is the electron density in the wall. If the dn electrons were to proceed undeflected, they would produce a number of ions dJ in the air of the cavity given by

$$dJ = dn \cdot F(R_w - x)t \quad (4)$$

where x is the distance travelled in the wall from the elementary volume $dx\,dy\,dz$ where the electrons originate to the edge of the cavity, $F(R_w - x)$ is the specific ionization in air at a distance $R_w - x$ from the end of the range, t is the distance travelled across the cavity, and R_w is the range of electrons of energy E_0 in the wall.

The total number of ions J (per quantum/cm²) in the cavity is

$$J = N_w \iint f(E_0)\,d\varphi \iiint_0^{R_w} F(R_w - x)t\,dx\,dy\,dz \cdot dE_0 \quad (5)$$

and it can be shown that the scattering of the electrons as they slow down in the wall does not change this integral (*8*).

The triple integral reduces to

$$V \int_0^{R_w} F(R_w - x)\,dx$$

where V is the volume of the cavity. The specific ionization, $F(R_w - x)$, can be written $-(dE/dx)_a/W$, where $-(dE/dx)_a$ is the specific energy loss in air at the energy E; W is the mean energy lost by an electron per ion pair produced and is substantially independent of energy. Thus

$$V \int_0^{R_w} F(R_w - x)\,dx = -V \int_{E_0}^{0} \frac{(dE/dx)_a}{W}\,dx = \frac{V}{W} \int_0^{E_0} \frac{(dE/dx)_a}{(dE/dx)_w}\,dE = \frac{V}{W} \int_0^{E_0} \frac{N_a S_a(E)}{N_w S_w(E)}\,dE \quad (6)$$

where $(dE/dx)_w$ is the specific energy loss in the wall material, N_a and N_w are the electron densities in the air and the wall, and $S_a(E)$ and $S_w(E)$ are the stopping powers per electron of the air and the wall. Substituting Eq. 6 in Eq. 5 and integrating with respect to φ gives

$$J = \frac{VN_a}{W} \int_0^{E_m} g(E_0) \int_0^{E_0} \frac{S_a(E)}{S_w(E)} \times dE \cdot dE_0 \quad (7)$$

where $g(E_0)dE_0$ is the probability per incident quantum/cm² that an electron will acquire an energy between E_0 and $E_0 + dE_0$ in any direction, and E_m is the maximum possible value of E_0.

In calculating the correction factor B', an expression for the stopping power of the wall material that did not include the finite-energy density effect was used. Calling this $S_w'(E)$, one can write the true stopping power as $S_w(E) = S_w'(E)\,[1 - \Delta]$, where Δ is the fractional reduction in the stopping power due to the finite-energy density effect. The ratio $S_a(E)/S_w(E)$ in Eq. 7 can therefore be replaced by $S_a(E)/S_w'(E)\,[1 - \Delta]$. Furthermore, the expression for the ionization obtained by neglecting the finite-energy density effect is

$$J' = \frac{VN_a}{W} \int_0^{E_m} g(E_0) \int_0^{E_0} \frac{S_a(E)}{S_w'(E)} \times dE \cdot dE_0$$

so that the fractional correction that must be applied to B' to take into account the finite-energy density effect, for γ-rays of a specific energy $h\nu$, is

$$D = (J - J')/J'$$

TABLE 1—Functions of I for Eq. 2

Substance	I(*ev*)	$2\ln I_w/I_a$	$(2\ln I_w/I_a)^2$
H (in solids)	15.6		
Be	60.4	−0.575	0.331
Graphite	76.4	−0.105	0.011
Air	80.5	0	0
O (in solids)	99.9		
Al	150	1.247	1.557
Anthracene	64.5*	−0.443	0.196
Water	68.9*	−0.312	0.097
Polystyrene	61.0*	−0.557	0.310
Lucite	65.3*	−0.418	0.175
Bakelite	68.8*	−0.314	0.098

* For compounds, I is the weighted geometric mean of the I's for the component elements.

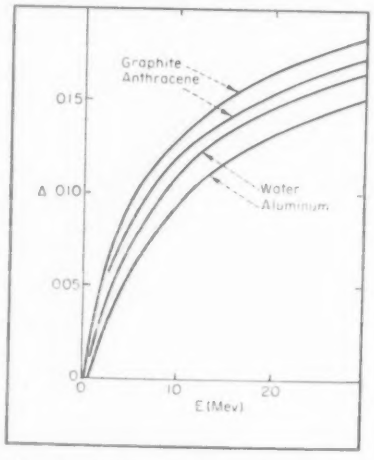

FIG. 2. Variation of fractional finite-energy density effect with electron energy

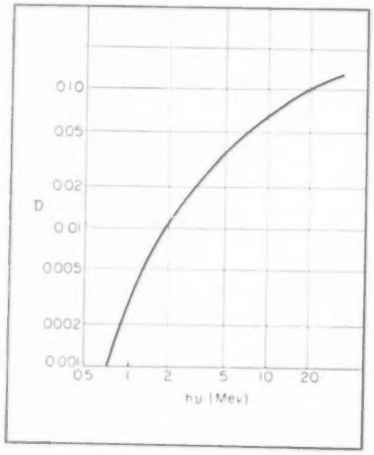

FIG. 3. Correction to measured dose in a graphite cavity, as function of γ energy

TABLE 2—Wall Correction Factors for the Gamma-Rays of Various Isotopes

	I^{131}			Au^{198}			Co^{60}			$Ra(B+C)$ (through 0.5-mm Pt)			Na^{24}		
	B′	D	B	B′	D	B	B′	D	B	B′	D	B	B′	D	B
Graphite	0.993	0	0.993	0.993	0	0.993	0.994	0.004	0.998	0.994	0.005	0.999	0.994	0.014	1.008
Polystyrene	0.961	0	0.961	0.962	0	0.962	0.967	0.004	0.971	0.966	0.005	0.971	0.969	0.014	0.983
Lucite	0.970	0	0.970	0.971	0	0.971	0.974	0.004	0.978	0.974	0.005	0.979	0.976	0.012	0.988
Bakelite	0.977	0	0.977	0.978	0	0.978	0.981	0.004	0.985	0.980	0.005	0.985	0.982	0.013	0.995
Aluminum	1.101	0	1.101	1.100	0	1.100	1.084	0	1.084	1.086	0	1.086	1.078	0.003	1.081

$$D = \int_0^{E_m} g(E_0) \int_0^{E_0} \left[\frac{\Delta}{1-\Delta}\right] \frac{S_a(E)}{S_w'(E)} \, dE \cdot dE_0 \Big/ \int_0^{E_m} g(E_0) \int_0^{E_0} \frac{S_a(E)}{S_w'(E)} \, dE \cdot dE_0 \quad (8)$$

This expression can be simplified by noting that the ratio $S_a(E)/S_w'(E)$ is very nearly constant over the range of energies of interest, so that finally, to a good approximation

$$D = \int_0^{E_m} g(E_0) \int_0^{E_0} \left[\frac{\Delta}{1-\Delta}\right] \times dE \cdot dE_0 \Big/ \int_0^{E_m} g(E_0) E_0 \, dE_0 \quad (9)$$

For a complex γ-ray spectrum, one must use a weighted mean of the values of D for each quantum energy.

Results

The values of Δ have been evaluated for graphite, anthracene, water, and aluminum from the recent paper of Sternheimer (*7*); Fig. 2 shows the variation of Δ with electron energy for these four substances. The values for graphite and anthracene differ by only about 10%, while those for water and aluminum are considerably lower. Since most cavity-chamber measurements have been made in chambers with graphite walls or walls differing little from carbon in atomic number, the remaining calculations have been carried out for graphite only.

The values of $\int_0^{E_0} [\Delta/(1-\Delta)]\, dE$ were obtained as a function of E_0 by graphical integration. Numerical values of the function $g(E_0)$ were next taken from the paper of Johns *et al.* (*12*), and with them the value of D was calculated as a function of $h\nu$, the incident γ-ray energy. The results of this calculation are shown in Fig. 3, where the quantity plotted is the fractional amount by which the factor B' given in Eq. 2 must be increased because of the finite-energy density effect.*

Full correction. The complete wall correction can now be written $B = B'(1 + D)$. B' is calculated from Eq. 2, and, for graphite, D can be taken directly from Fig. 3.

Approximate values of D for some other wall materials can be obtained with the aid of Fig. 2. For example, polystyrene is quite close to anthracene in atomic composition, so that by estimating from Fig. 2 the amount by which Δ for anthracene differs from Δ for graphite over the range of electron energies involved, one can make a useful guess at the appropriate value of D for anthracene and hence for polystyrene. Thus for 20-Mev γ-rays, the mean initial energy of the Compton electrons is roughly 10 Mev, and Fig. 2 shows that in the region between 0 and 10 Mev, Δ for anthracene is about 10% less than Δ for graphite. Assuming the same result for polystyrene, one finds from Fig. 3 that D should be about $0.105 \times 0.9 = 0.095$. For the much lower γ-ray energies that are generally emitted by radioisotopes, where D itself is not more than 1 or 2%, this procedure should be accurate enough for most purposes.

In Table 2, the values of B', D and B are presented for a few common gamma emitters and wall materials. They are about 2% higher than the values given by Laurence, corresponding to a *reduction* of 2% in the corresponding corrected dose rates.

Radium output. The results for Ra(B + C) γ-rays are of particular interest, since they bear on the standard figure for the absolute γ-ray output of radium. The results of a number of different cavity-chamber measurements of the output of radium in 0.5 mm of platinum have recently been compiled and corrected in a uniform manner for wall effect (*13*) on the basis of Laurence's calculations; they then yield a mean value of 8.41 r/mg/hr at 1 cm. When the correction factors given in Table 2 are used instead, the figure becomes 8.2 r/mg/hr at 1 cm. Thus the apparent discrepancy between the cavity-chamber values of about 8.4 and Taylor and Singer's (*14*) free-air value of 8.16 r/mg/hr at 1 cm can probably be attributed to the influence of the density effect on the cavity-chamber measurements.

* * *

It is a pleasure to acknowledge the assistance of Miss Catherine May with the numerical work, and to thank Dr. R. M. Sternheimer for a helpful exchange of letters.

* Since the completion of these calculations, the very precise stopping-power measurements of T. J. Thompson have become available (UCRL-1910, 1952). Although the values of I he obtains are mostly lower than those of Bakker and Segrè, the corresponding *relative* stopping powers are very nearly the same. In fact, if one includes the zero-energy density effect of 1% in liquid air calculated by Sternheimer (*Phys. Rev.*, in press), Thompson's results for both graphite and polystyrene lead to wall corrections differing by only a few tenths of a percent from those given in Table 2. I am grateful to several members of the Radiation Physics Laboratory of the National Bureau of Standards for a helpful discussion of this point.

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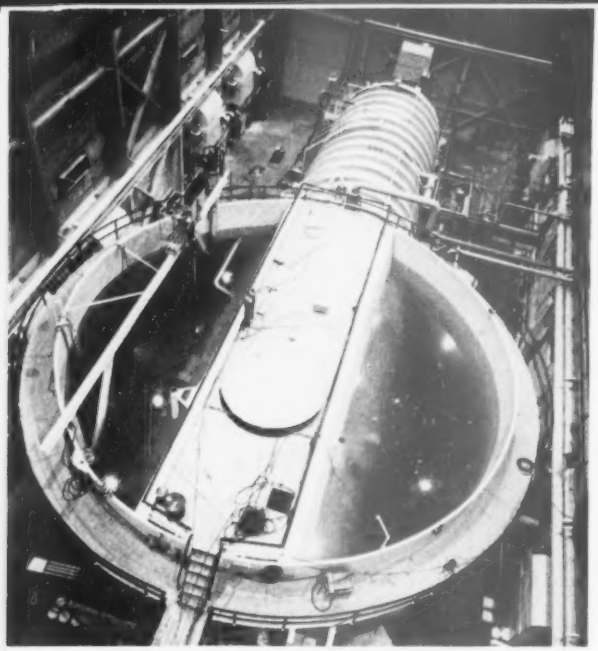
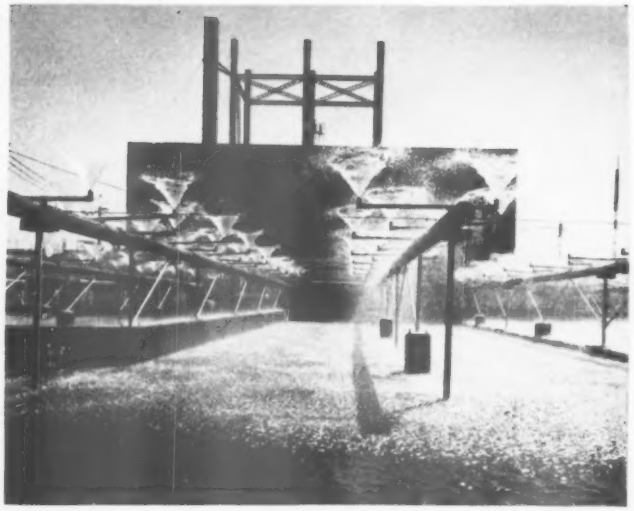


STR PROTOTYPE operating inside mock-up of submarine hull produces power dissipated in . . .



SPRAY POND holding 2,000,000 gallons of water. Useful application of nuclear power is near as . . .

PIT, 25-FT DEEP, when filled with water, holds "hot" reactor for disassembly. Smaller pieces are placed on elevator (rear) and hoisted to hot cell.

Nautilus Awaits Reactor

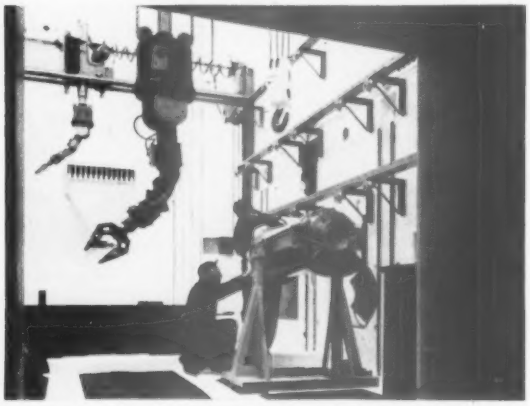
PICTURES SHOW the Idaho facilities used by Westinghouse Electric Corporation for testing the Submarine Thermal Reactor prototype (Mark I).

Launched January 21, the *Nautilus* is now ready to have the actual STR (Mark II) installed. This, the first nuclear-propelled vessel, will provide a proving ground for the first practical application of nuclear power.

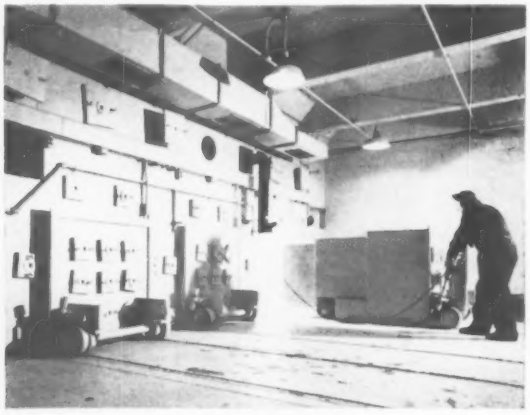
Actually, the STR prototype has been producing experimental power for about nine months using the facilities shown in these pictures. The Mark I reactor is under the hatch near the center of the tank shown above (left). Tank contains 285,000 gallons of water, is about 50-ft wide, 40-ft high.

Building housing reactor prototype is shown behind spray pond (right). Pond has 288 nozzles and can cool 22,500 gal/min.

In emergency, fans can exhaust air from building at rate of 36,000 ft^3/min from each of six stacks.



INTERIOR OF HOT CELL used for work on radioactive components. Here, well-ventilated hot cell is being used for manual work on Mark I component before installation on power plant



HOT CELL HAS six 4-ton doors, each 4-ft square at large end, 7-ft thick. A seventh door weighing 25 tons accommodates very large components

Multigroup Methods for Neutron Diffusion Problems

An essential factor in reactor design is the neutron flux space-energy distribution. For a nonthermal reactor, this spectrum can be calculated by dividing the energy range into discrete intervals characterized by distinct spatial flux distributions. Methods for doing this are developed and applied to an actual analysis

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THE AGE-DIFFUSION equation provides a good description of the neutron behavior of a large class of nonthermal reactors. Thus, a convenient means of obtaining an accurate solution to this equation is very useful for reactor design calculations.

One of the most convenient ways of mathematically handling this equation is the multigroup method,† in which the relevant energy range is divided into several discrete intervals. Thus the age-diffusion equation is reduced to a finite set of coupled ordinary differential equations called multigroup equations. Neutrons in each energy interval are treated as a separate group.

Analytical and numerical methods for solving multigroup equations with an assumed source are discussed in this article. The adjoint equation is also reduced to multigroup form, and the use of the adjoint function to obtain improved reactivity values is described.

Assemblies that are large compared with the neutron mean free path and in which the neutron energy is reduced primarily by elastic collisions with the atoms in the assembly will be considered here. If the atoms in the assembly all have mass substantially greater than the neutron mass, the Boltzmann integral equation for neutron diffusion may be converted into a diffusion-type differential equation‡ known as the age-diffusion equation

$$-\nabla \frac{1}{3\sigma_{tr}} \nabla nv + \sigma_a nv = -\frac{\partial q}{\partial u} + S(r,u)$$

$$-\nabla \frac{1}{3\sigma_{tr_T}} \nabla nv_T + \sigma_{a_T} nv_T = q_T \quad (1)$$

where§ q = slowing down density = $nv\xi\sigma_s$, σ_{tr} = *macroscopic* transport cross section, σ_s = *macroscopic* moderator scattering cross section, σ_a = *macroscopic* absorption cross section (includes inelastic scattering), u = lethargy = $\ln 10^7/E$ where E is in ev, nv = neutron flux per unit u, and ξ = average logarithmic energy loss in a moderating collision. The subscript T indicates the thermal group. The source term $S(r,u)$ represents neutrons liberated in fissions ($S_f(r,u)$) and neutrons produced at u by inelastic scattering from a higher energy ($S_{in}(r,u)$). The fission contribution may be written

$$S_f(r,u) = \nu \left[\int_{-\infty}^{u_T} nv\,\sigma_f\,du' + nv_T\,\sigma_{f_T} \right] \chi(u)$$

where ν is the number of neutrons emitted per fission and $\chi(u)$ is the fission spectrum normalized so that $\int \chi(u)\,du = 1$.

The inelastic scattering contribution may be written

$$S_{in}(r,u) = \int_{-\infty}^{u} nv\,\sigma_{in}(u')\,\chi(u' \to u)\,du'$$

The fraction of neutrons scattered inelastically at energy u' that are degraded to energy u is $\chi(u' \to u)$ and $\int_{-\infty}^{\infty} \chi(u' \to u)\,du = 1$.

The most straightforward procedure for numerically integrating Eq. 1 would appear to be to calculate $\partial q/\partial u$ as a function of position from Eq. 1 for a given value of u and to use this derivative to obtain the flux at the next value

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* Operated for the Atomic Energy Commission by the General Electric Company.

† The particular formulation of the multigroup method discussed here stems from a classified KAPL report issued in May, 1947. A general consideration of multigroup methods can be found in the book by S. Glasstone, M. C. Edlund, "The Elements of Nuclear Reactor Theory," p. 227 ff (D. Van Nostrand Co., Inc., New York, 1952). Among other techniques are the various mode methods such as those developed by H. Brooks and L. Tonks (KAPL), and G. Goertzel (NDA); the numerical method of A. S. Thompson, *J. Appl. Phys.* **22**, 1223 (1951); the multicell method of G. Safonov (Rand), or direct integration with respect to energy. We have found direct integration with respect to energy useful in cases where the flux is attenuated very rapidly. Thompson's method is equivalent to scheme A (see p. 24) with $f = 1$ and uses a different method for integrating the spatial equation numerically. Integral theory methods have been derived by R. P. Feynman and T. A. Welton at Los Alamos and by G. Safonov. The spherical harmonic method has also been used extensively for treating energy dependent transport problems in which diffusion theory is not valid.

‡ See R. E. Marshak, H. Brooks, and H. Hurwitz, Jr., "Introduction to the Theory of Diffusion and Slowing Down of Neutrons," NUCLEONICS **4**, No. 5, 11 (1949); **4**, No. 6, 43 (1949); **5**, No. 1, 53 (1949); **5**, No. 2, 59, (1949).

§ More accurate results in situations where σ_a is not small compared to σ_s may be obtained by replacing $\xi\sigma_s$ by a linear function of σ_s and σ_a. We have adopted the alternative approach of correcting the absorption cross sections to account for the effects of spatial and energy self-shielding. In Eq. 1, q_T is evaluated at an energy just above thermal.

of u. This procedure is not stable unless the intervals in u are small relative to the spatial intervals. Since rather small spatial intervals are necessary to obtain an adequate description of the spatial flux variation, it is necessary in this method to use an undesirably small increment of u. Although this "direct integration" method has occasionally been used for studying the low energy part of the neutron spectrum, it has generally been found more convenient to obtain the numerical solution of Eq. 1 by the methods discussed in the following section.

MULTIGROUP EQUATION

It has been customary at KAPL to work with Eq. 1 expressed in terms of the slowing down density, q, rather than the flux, $n\nu$. Since $q(u)$ represents the number of neutrons being scattered through u per unit time and volume, it follows that if leakage and absorption effects are small, $q(u)$ will be a slowly varying function of u. If σ_s varies with energy, $q(u)$ tends to be a less sensitive function of energy than the flux, $n\nu$, which will be roughly proportional to $1/\sigma_s$. For this reason, it is preferable to perform the required averaging of Eq. 1 over the individual energy groups in terms of the variable q.

In terms of q, Eq. 1 becomes

$$-\frac{1}{3\xi\sigma_s\sigma_{tr}}\nabla^2 q+\frac{\sigma_a}{\xi\sigma_s}q = -\frac{\partial q}{\partial u}+S(r,u) \quad (2)$$

It has been assumed that cross sections are constant in a given spatial region so that the diffusion coefficient can be brought in front of the divergence operator.

To go from Eq. 2, in which u is a continuous variable, to the group equations, Eq. 2 is integrated with respect to u over the logarithmic energy range U_i corresponding to a particular group, i.

We then obtain the equation

$$-U_i\left(\frac{1}{3\xi\sigma_s\sigma_{tr}}\nabla^2 q\right)_{i_{avg}}+U_i\left(\frac{\sigma_a}{\xi\sigma_s}q\right)_{i_{avg}} = -q_i{}^O+q_i{}^I+\int_{U_i}S(r,u)\,du \quad (3)$$

The averages are with respect to u in the group. The superscripts O and I indicate values of q at the high u and low u end of the group, respectively. Thus, $q_i{}^I$ is the degradation into the ith group from the previous group and $q_i{}^O$ is the degradation out of the ith group. The approximation of replacing the averages of products appearing in Eq. 3 by the product of averages yields the following

$$-\left(\frac{U}{3\xi\sigma_s\sigma_{tr}}\right)_{i_{avg}}\nabla^2(q)_{i_{avg}}+\left(\frac{U\sigma_a}{\xi\sigma_s}\right)_{i_{avg}}(q)_{i_{avg}}+q_i{}^O = q_i{}^I+\int_{U_i}S(r,u)\,du \quad (4)$$

This is justified if either q or the cross sections do not change radically over the group. Hence, this is a criterion which must be considered in choosing the energy subdivisions for the groups. For the purpose of analysis of experiments, it is convenient to have group divisions occur at energy values where there is some physical threshold or discontinuity such as the Cd-cutoff energy.

To solve Eq. 4 when $q_i{}^I$ and S are known, it is necessary to postulate an additional relationship between (q), q^O, and q^I. This relationship is taken to be of the form

$$(q)_{avg}=\omega_1 q^I+\omega_2 q^O \quad (5)$$

where ω_1 and ω_2 are appropriately chosen constants.

Since $q_{i-1}{}^O = q_i{}^I$, from Eq. 5 is obtained

$$q_i{}^O=\left(\frac{q}{\omega_2}\right)_{i_{avg}}-\frac{\omega_1}{\omega_2}q_i{}^I = \left(\frac{q}{\omega_2}\right)_{i_{avg}}-\frac{\omega_1}{\omega_2}q_{i-1}{}^O \quad (6)$$

Hence, the group equation of Eq. 4 becomes

$$-\left(\frac{U}{3\xi\sigma_s\sigma_{tr}}\right)_{i_{avg}}\nabla^2(q)_{i_{avg}}+\left(\left[\frac{\sigma_a U}{\xi\sigma_s}\right]_{i_{avg}}+\frac{1}{\omega_2}\right)(q)_{i_{avg}} = \frac{\omega_1+\omega_2}{\omega_2}q_{i-1}{}^O+\int_{U_i}S\,du \quad (7)$$

The diffusion theory boundary conditions have been used at interfaces between different media. Thus, the group average neutron density

$$\left(\frac{1}{\xi\sigma_s}\right)_{i_{avg}}(q)_{i_{avg}}$$

and current

$$\left(\frac{1}{3\xi\sigma_s\sigma_{tr}}\right)_{i_{avg}}\nabla(q)_{i_{avg}}$$

are made continuous.

Equation 4 can also be written

$$-\nabla D_i\nabla NV_i+\Sigma_{a_i}NV_i = q_{i-1}{}^O-q_i{}^O+\int S\,du \quad (8)$$

where

$$D_i=\frac{\left(\frac{1}{3\xi\sigma_s\sigma_{tr}}\right)_{i_{avg}}}{\left(\frac{1}{\xi\sigma_s}\right)_{i_{avg}}}$$

$$NV_i=U_i\left(\frac{1}{\xi\sigma_a}\right)_{i_{avg}}(q)_{i_{avg}}$$

$$\Sigma_{a_i}=\frac{\left(\frac{\sigma_a}{\xi\sigma_s}\right)_{i_{avg}}}{\left(\frac{1}{\xi\sigma_s}\right)_{i_{avg}}}$$

In this equation, NV_i and $D_i\nabla NV_i$ are continuous at all interfaces. Using Eq. 5

$$q_i{}^O=\frac{1}{\omega_2}\left(\frac{NV_i}{U_i\left(\frac{1}{\xi\sigma_s}\right)_{i_{avg}}}-\omega_1 q_{i-1}{}^O\right)$$

so that Eq. 8 becomes

$$-\nabla D_i\nabla NV_i+\left(\Sigma_a+\frac{1}{\omega_2 U_i\left(\frac{1}{\xi\sigma_s}\right)_{i_{avg}}}\right)NV_i = \left(1+\frac{\omega_1}{\omega_2}\right)q_{i-1}{}^O+\int S\,du \quad (9)$$

Particular Schemes

Several choices of ω_1 and ω_2 have been used in practice. One choice, which is referred to as scheme A, corresponds to setting ω_1 equal to zero and ω_2 equal to f, a number chosen by trial-and-error for each group and region of the assembly so that f times the spatial average of q^O gives a reasonable value for the spatial average of $(q)_{avg}$. The solution is not sensitive to the choices of f and it is usually possible, on the basis of experience, to make adequate choices the first time. Ordinarily, f will be somewhat greater than unity in a group and region not containing a neutron source but may be less than unity in groups and regions in which sources occur.

In scheme B, ω_1 and ω_2 are set equal to 1/2. This eliminates the guesswork of scheme A, but, in certain situations it has the disadvantages discussed below. At Oak Ridge, a modification of scheme B has been used in which a linear relationship is assumed to hold for the flux rather than the slowing down density. Thus*

$$(nv)_{avg} = \frac{1}{2}(nv^I + nv^O)$$

$$= \left(\frac{1}{\xi\sigma_s}\right)_{avg} (q)_{avg} \quad (10)$$

Since

$$q^O = (\xi\sigma_s)^O nv^O; \; q^I = (\xi\sigma_s)^I nv^I \quad (11)$$

Equation 10 corresponds to

$$\omega_1 = (1/\xi\sigma_s)^I/(2/\xi\sigma_s)_{avg} \quad (12)$$
$$\omega_2 = (1/\xi\sigma_s)^O/(2/\xi\sigma_s)_{avg}$$

In this scheme, nv^O and nv^I as well as nv_{avg} are continuous across interfaces. This tends to reduce oscillations at boundaries between highly dissimilar regions.

Accuracy of Method

The accuracy of the various schemes may be investigated by comparison with the age-theory result for a bare reactor (or a single spatial Fourier component) in which case $\nabla^2 q = -k^2 q$, where k is independent of energy.

If $S(r,u) = 0$, Eq. 2 yields

$$q_i^O = q_{i-1}^O e^{-x} + e^{-x} \int_{U_i} du' \, S(r,u') \times$$
$$\exp\left\{\int^{U'} \frac{\sigma_a}{\xi\sigma_s} du'' + \int^{U'} \frac{k^2}{3\xi\sigma_s\sigma_{tr}} du''\right\} \quad (13)$$

where

$$x = \int_{U_i} \frac{\sigma_a}{\xi\sigma_s} du + \int_{U_i} \frac{k^2}{3\xi\sigma_s\sigma_{tr}} du \quad (14)$$

Equation 7 has the solution

$$q_i^O = q_{i-1}^O \frac{1-\omega_1 x}{1+\omega_2 x} + \frac{1}{1+\omega_2 x} \int_{U_i} S \, du \quad (15)$$

Hence, if the source term is neglected, a comparison of Eq. 13 and Eq. 15 shows that the assumption of Eq. 5 is equivalent to replacing e^{-x} by $(1-\omega_1 x)/(1+\omega_2 x)$. For small x, the expansions agree only if $\omega_1 + \omega_2 = 1$. Scheme B with $\omega_1 = \omega_2 = 1/2$ gives the best representation for small and intermediate values of x since the quadratic terms in the expansion also agree. Unfortunately, $(1-(x/2))/(1+(x/2))$ goes to -1 for large x instead of to zero as does e^{-x}. Thus, in scheme B the higher modes that correspond to large k, and hence large x, are treated incorrectly. This leads to oscillations in q from group to group and, in certain cases, even to negative values for q. If the group widths are taken sufficiently small, the oscillatory troubles may be unimportant. Thus, scheme B is particularly suitable for calculations in which a large number of groups are used and high accuracy is desired. For calculations with a comparatively small number of groups, scheme A is preferred. If it happens that the assumed value of f satisfies the consistency requirement

$$f(q)_{avg} dV = f \int q^O \, dV = \tfrac{1}{2} f(q^O + q^I) \, dV$$

then, as far as the fundamental mode is concerned, scheme A is effectively equivalent to scheme B. For the higher modes (large x), $1/(1+fx)$ approaches zero, so that even though a good approximation to e^{-x} is not obtained for these modes, they are damped sufficiently to prevent oscillations.

The Oak Ridge modification of scheme B will be accurate for small and intermediate values of x provided that $\xi\sigma_s$ does not vary appreciably over the energy range of the group. Then, from Eq. 12, ω_1 and ω_2 will both be close to $1/2$. The condition $\omega_1 + \omega_2 = 1$ will, in general, not be satisfied exactly in this scheme, but in practice this will not introduce serious errors.

All the preceding schemes have the common property of conserving neutrons. That is, for each region R

$$\int_R q_i^I \, d(vol) - \int_R q_i^O \, d(vol)$$
$$+ \int_R \int_{U_i} S \, du \, d(vol) = \Sigma_{a_i} \int_R N V_i \, d(vol)$$
$$- D_i \int_R \nabla N V_i \, d(surface) \quad (16)$$

This is obtained by integrating Eq. 8 over the volume of a region, R. Since neutron current is continuous at all interfaces, the conservation law holds for the assembly as a whole.

Closely related to the conservation law is the fact that the multigroup equations reduce to the proper one-velocity equation if the absorption and transport cross sections are independent of energy. This is the basic reason for the good reactivity results that can be obtained with a comparatively small number of groups. Even though neutrons may, in effect, be assigned to slightly incorrect energies, no neutrons are actually lost from the accounting. Since the cross sections are, in general, not sensitive functions of energy, the errors in spectrum do not lead to correspondingly large errors in reactivity.

Adjoint Equation

The adjoint functions are useful in first-order perturbation calculations involving reactor materials and cross sections and in the interpretation of many reactor experiments.† Since the multigroup or age-diffusion equations are not self-adjoint, a separate solution must be obtained either by writing the system adjoint to the age-diffusion Eq. 1 and then deriving multigroup adjoint equations in the usual way or by writing down the multigroup system adjoint to the multigroup Eq. 7. These two methods of approach lead to adjoint systems of slightly different form. We prefer the former method since, in this case, the physical relationship of the adjoint function at the high and low energy ends of the group to the group average is preserved, while it is not in the latter case.

The system adjoint to Eq. 1 is

$$-\nabla \frac{1}{3\sigma_{tr}} \nabla m + \sigma_a m$$
$$= \xi\sigma_s \frac{\partial m}{\partial u} + \nu\sigma_f \int \chi m \, du \quad (17)$$
$$-\nabla \frac{1}{3\sigma_{tr_T}} \nabla m_T + \sigma_{a_T} m_T = \sigma_{f_T} \nu \int \chi m \, du$$

The energy dependence of the source term in Eq. 17 is determined by that of the fission cross section, σ_f. The spatial dependence is determined by $\int \chi(u) m(r,u) \, du$. This integral is proportional to the iterated fission probability at point r, averaged over the fission spectrum, i.e., to $F_H(r)$. If $m(r,u)$ is normalized to be equal to the iterated fission probability $F(r,u)$,‡ then $\int \chi \, m \, du = F_H$. The function $F_H(r)$ plays the same role in the adjoint problem as does the power distribution, $\int nv\sigma_f \, du$, in the original problem. In fact, from the viewpoint of the section on analytical solutions, F_H is adjoint to the power distribution. Just as, in the flux equations, a spatial variation

* In the Oak Ridge formulation developed by D. K. Holmes and associates, σ_s is taken to include the absorption cross section as well as the scattering cross section.

† The use of the adjoint equation in neutron diffusion problems is due to E. P. Wigner.

‡ The function $F(r,u)$, proportional to m, is defined by the relation

$$F(r,u) = \frac{m \int\int nv\sigma_f \, du \, d(vol)}{\int\int [\nu nv\sigma_f \int \chi m \, du'] \, du \, d(vol)}$$

and is called the iterated fission probability. It can be shown that $F(r,u)$ represents the number of fissions that occur per generation a large number of generations after a neutron of lethargy u has been inserted into the assembly at point r.

for the power distribution may be assumed and the multigroup equations may be used to calculate the resulting power distribution, so may a spatial distribution be assumed for $F_u(r)$ and the adjoint equations used to calculate the resulting $F_u(r)$. As in the original flux equation, the correct source variation with position is determined by the consistency requirement between the assumed and calculated source. Since the adjoint function does not have the peaks near the reflector that often occur in the power distribution, it is somewhat easier to guess the correct shape for F_u than for the power distribution.

If Eq. 17 is divided by $\xi\sigma_s$ and integrated over a lethargy group, the following equation is obtained

$$-\left(\frac{U}{3\xi\sigma_s\sigma_{tr}}\right)_{i_{avg}}\nabla^2(m)_{i_{avg}} + \left(\frac{\sigma_s U}{\xi\sigma_s}\right)_{i_{avg}} m_i = m_i{}^I - m_i{}^O + \left(\frac{\sigma_f U}{\xi\sigma_s}\right)_{i_{avg}} \nu \sum_j \chi_j m_{j_{avg}} \quad (18)$$

The superscripts I and O refer here to the low energy (high u) and high energy (low u) ends of the group. This is opposite to the usage in the case of the flux equation since in the adjoint equation the "degradation" goes from low energy to high energy because of the change in sign of the lethargy derivative term. The boundary conditions at interfaces are that $(m_i)_{avg}$ and

$$\left[\left(\frac{1}{3\xi\sigma_s\sigma_{tr}}\right)_{i_{avg}} \Big/ \left(\frac{1}{\xi\sigma_s}\right)_{i_{avg}}\right]\nabla(m_i)_{avg}$$

are continuous. These follow from the boundary conditions on Eq. 17 that m and $(1/3\sigma_{tr})\,\nabla m$ are continuous. The above multigroup equations may be solved in the usual way by assuming, for example, that $(m_i)_{avg} = (1/2)(m_i{}^I + m_i{}^O)$. An objectionable feature of the equations in this form is that no simple conservation law applies. It is possible to derive conservation laws for individual regions, but if $(1/\xi\sigma_s)_{avg}$ is not continuous, the current is not continuous across boundaries. This makes it difficult to check the correctness of numerical solutions. The failure of conservation to hold is a property of the original adjoint equation. It arises from the fact that $\xi\sigma_s$ occurs outside the derivative of m with respect to u.

It is possible to write Eq. 17 in a form suitable for obtaining a conservation law by introducing the quantity $M = \xi\sigma_s m$. Then Eq. 17 becomes

$$-\nabla\frac{1}{3\sigma_{tr}}\nabla\frac{M}{\xi\sigma_s} + \left(\frac{\sigma_a}{\xi\sigma_s} + \frac{\partial \ln \xi\sigma_s}{\partial u}\right)M = \frac{\partial M}{\partial u} + \sigma_f\nu\int\chi\frac{M}{\xi\sigma_s}\,du' \quad (19)$$

This equation has essentially the same form as Eq. 2 with q replaced by M and with a virtual absorption cross section $\partial\xi\sigma_s/\partial u$ added to the actual absorption cross section σ_a. Hence, a conservation law that is analogous to the conservation of neutrons may be derived with nv replaced by m and q replaced by M. In addition, the virtual absorption cross section must be included with the actual cross section in computing the equivalent absorption to be used in the conservation relationship.

It should be noted that in the adjoint multigroup equation corresponding to Eq. 17 or 19 there is a source term in all groups since σ_f is finite at all energies. In the high energy groups, however, the source term is comparatively unimportant since σ_f is small. The adjoint function does not vanish at any finite value of the energy, but becomes small as the energy increases, corresponding to the small probability that an extremely fast neutron will not escape from the assembly. The adjoint calculation may be carried to as high an energy as desired, but for practical purposes, it is not necessary to carry the calculations to energies higher than the important range of the fission spectrum.

Inelastic scattering may be included in the adjoint multigroup equations in much the same manner as in the flux equations. The inelastic scattering introduces a source term in the equation for the ith group that is proportional to the sum of the cross sections for inelastic scattering from the ith group to each of the lower energy groups times the adjoint functions in the corresponding groups.

Since the eigenvalue, ν_c, in the age-diffusion equation (Eq. 1) and the adjoint equation (Eq. 17) must be the same, the multigroup approximations to these equations will have approximately the same eigenvalues. In practice, the agreement between ν_c as obtained from the two multigroup equations has been satisfactory.

METHODS OF SOLUTION

The number of energy groups and spatial regions for which a solution is desired influences the choice of methods. The availability of calculating facilities also has a bearing on the choice.

Analytical Methods

In principle, Eqs. 7 and 18 can be solved analytically since they form a set of coupled linear differential equations. However, for more than two or three groups or two or three spatial regions, such a solution becomes extremely cumbersome because of the large order determinants involved. In such cases an approximation method is desirable. Our methods are based on assuming a source distribution for fission neutrons and calculating the resulting neutron flux in each group successively, starting from the highest energy group.* From the calculated fluxes a new fission distribution is obtained. The ratio of the source strength to the number of fissions produced is then a first approximation to ν_c, which is defined as that value of ν that would make the assembly just critical.

The ratio of the actual value of ν to ν_c gives the reactivity of the assembly. The process can be repeated, starting with the new fission distribution as the source, until self-consistency (identical assumed and calculated fission distributions) is attained.

The accuracy of the value of ν_c obtained at a particular stage of the iteration process (with a particular assumed source shape) can be improved by using adjoint theory. For a specific assembly, one can define the function $P(r' \to r)$, which gives the number of fissions per unit volume at point r due to a fission neutron inserted in the assembly at point r' (r and r' are here considered to be vectors). The quantity ν_c is the eigenvalue of the integral equation

$$(1/\nu_c)\,N(r) = \int N(r')P(r' \to r)\,dV'$$

Here $N(r)$ is the number of fissions per unit volume. All volume integrations are understood to be over the core.

* In the case of the adjoint equations, the procedure is similar except that the spatial dependence of the fission neutron adjoint function is assumed and then the equations are solved starting with the thermal group and working upward in energy. For the group i_0 next to the thermal group, $m_{i_0}{}^I$ is set equal to m_T.

The adjoint equation has the form

$$\frac{1}{\nu_c} M(r) = \int M(r')P(r \rightarrow r')\, dV'$$

The adjoint function $M(r)$ appearing in this equation is proportional to the adjoint function $m(r,u)$ defined in Eq. 17 averaged over the fission spectrum.† Thus, it is proportional to F_H, the iterated fission probability for fission neutrons as defined in connection with the adjoint equation. If the integral equation for N is multiplied by M and integrated over the core, we obtain

$$\frac{1}{\nu_c} = \frac{\iint N(r)P(r \rightarrow r')M(r')\, dV\, dV'}{\int N(r)M(r)\, dV}$$

If slightly incorrect functions are inserted for N and M, the error in ν_c will be of the order of magnitude of the product of the errors in N and M. In particular, if the correct function is used for either N or M, the corresponding value of ν_c will be correct regardless of the error in the other function. In applying this expression for ν_c, it is not necessary to work explicitly with the kernel $P(r' \rightarrow r)$. If one assumes for the spatial distribution of fission neutrons the function $N_0(r)$,‡ then the function N_1 defined by

$$N_1 = \int N_0(r')P(r' \rightarrow r)\, dV'$$

represents the fission distribution resulting from the assumed source N_0. The function N_1 may, therefore, be calculated directly by the multigroup method by using for $S(r,u)$ in Eq. 1 the function $\chi(u)N_0(r)$. If $M(r)$ is approximated by the function M_0, the expression for ν_c becomes

$$\frac{1}{\nu_c} = \frac{\int M_0 N_1\, dV}{\int M_0 N_0\, dV} \qquad (19a)$$

Note that the normalizations of N_0 and M_0 are irrelevant in this expression since if N_0 is multiplied by a constant factor, N_1 will be multiplied by the same factor. If the very crude assumption is made that M_0 is independent of position, the above relation gives ν_c as the ratio of total neutrons in the assumed source divided by the total number of fissions resulting from this source.§ In practice, it is possible to make a considerably better assumption for M_0. On the basis of adjoint function calculations, it appears that M can often be fairly well approximated by $\cos kr$, where $k \cong \pi/4r_0$; r_0 is the core radius. Another approximation occasionally employed is $M_0 = 1 - (\alpha r^2/r_0^2)$, where $\alpha \cong 0.5$. The insensitivity of ν_c to the form of M_0 is an indication of the adequacy of the choice of N_0.

A further increase in the accuracy of ν_c can be obtained by calculating an improved approximation M_1 for M by using the initial assumption M_0 as a source in the adjoint multigroup equation (Eq. 17). M_1 will then bear the same relation to M_0 as N_1 bears to N_0. Thus,

$$M_1(r) = \int P(r \rightarrow r')M_0(r')\, dV'$$

It is interesting to note that replacing M_0 by M_1 in the equation for ν_c gives exactly the same result as assuming N_1 for the source in the multigroup equations, calculating the resulting fission distribution N_2 and using M_0 for the adjoint function. Thus

$$\frac{1}{\nu_c} \cong \frac{\int M_0 N_2\, dV}{\int M_0 N_1\, dV} = \frac{\iint N_1 P M_0\, dV\, dV'}{\iint N_0 P M_0\, dV\, dV'} = \frac{\int N_1 M_1\, dV}{\int N_0 M_1\, dV}$$

In practice, it is often preferable to calculate M_1 instead of N_2 (to iterate the adjoint function once and the fission distribution once rather than to iterate the fission distribution twice). This is because it is often desirable to know the adjoint functions as accurately as the fission distribution. Whereas N_2 is usually not a great improvement over N_1, M_1 may be a substantial improvement over M_0.

In the following sections specific methods will be discussed for using the multigroup equations to obtain N_1 from N_0 or M_1 from M_0.

Semianalytical Methods

This technique is based on assuming a sufficiently simple analytical expression for the source S that the individual group equations can be solved analytically. If the source is expressed as a linear combination of eigenfunctions of the Laplacian operator, then it is possible to calculate the entire neutron distribution group by group. The slowing down densities in each group and region will also be linear combinations of eigenfunctions of the Laplacian operator. The solution for $(q_i)_{s+g}$ in Eq. 7 for a particular region and group will be a linear combination of the functions that appear on the right-hand side of the equation (i.e., those functions that have been introduced by the source and the previous groups) plus the two homogeneous solutions of the left-hand side. The coefficients of the homogeneous solutions are determined by the boundary conditions at interfaces and at the outer surface of the reactor. In this way, it is possible to solve the equation for each group in succession, starting with the highest energy group.

It is convenient to take for the source function a linear combination of the first few members of an orthonormal set of functions. If J functions are used in the expression for the source, the resulting fission distribution can be calculated from the multigroup equations in terms of the J arbitrary coefficients in the source expression. It is then possible to obtain J homogeneous linear equations for these coefficients by requiring that the coefficients of the first J terms in the expansion of the calculated fission distribution are $1/\nu_c$ times the corresponding coefficients in the source expansion. The requirement that these J equations have a solution determines ν_c, and the corresponding solution for the coefficients determines the best representation of the source in terms of the J functions. This procedure is equivalent to writing

$$N = \sum_{i=1}^{J} \alpha_i \phi_i$$

$$M = \sum_{i=1}^{J} \beta_i \phi_i$$

where ϕ_i represents a member of the orthonormal set, and calculating

$$\frac{1}{\nu_c} = \frac{\iint MPN\, dV\, dV'}{\int MN\, dV} = \frac{\sum_{i,j} \beta_i P_{ij} \alpha_j}{\sum_i \alpha_i \beta_i}$$

where

$$P_{ij} = \iint \phi_i(r)P(r \rightarrow r')\phi_j(r')\, dV\, dV'$$

If it is required that the variation of this expression for $1/\nu_c$ be zero for variations in both α_i and β_i, linear homogeneous equations will be obtained for both α_i and β_i. The equations for the α_i's will be the same as before and so lead to the same value of ν_c. Furthermore, the equations for the β_i's can be solved and will provide the best rep-

† It is not the same as the function $M(r,u)$ used in Eq. 19.

‡ Here N_0 may represent the initial assumption for the fission distribution or may be the result of one or more iterations of the initial assumption.

§ This is the value of ν_c obtained without making use of adjoint theory.

resentation for the adjoint function in terms of the first J functions of the orthonormal set.*

The above considerations indicate that the number, J, of functions employed in the expression for the source does not have to be large to obtain an accurate value for ν_c. In practice, good results may be obtained with J equal to two or three.

The principle difficulty with the analytical method is that in problems involving a large number of groups or regions, the number of functions that are introduced is large so that the calculation of the lower energy groups becomes quite tedious. Furthermore, considerable cancellation occurs in the numerical calculations so that these calculations must be done with extreme care. However, for two-region problems with four or five groups, the method is quite feasible.

Numerical Methods

When the number of groups is large or the number of spatial regions is greater than two, it is simpler to solve Eq. 9 by numerical methods. The spatial integration of the differential group equation of Eq. 9 can be performed numerically for a reactor with spherical symmetry in the following way. Equation 9 is first simplified by the substitutions

$$F^i = rNV_0Q^{i\prime} = rq_i{}^o, \quad \text{and}$$

$$S_i = r\int_{U_i} S\,du$$

Then Eq. 9 becomes

$$-D_i\frac{d^2F_i}{dr^2} + \left[\Sigma_{a_i} + \frac{1}{\omega_2 l^i{}_i\left(\frac{1}{\xi\sigma_s}\right)_{i_{avg}}}\right]F_i = \left(1+\frac{\omega_1}{\omega_2}\right)_i Q^{i-1\prime} + S^i \quad (20)$$

Then, divide the spatial region into a set of subintervals of width Δr and let the index n refer to the radius r_n at the nth space point. Using the difference approximation,†

$$\left(\frac{d^2F^i}{dr^2}\right)_{r_n} = \frac{F_{n+1}{}^i + F_{n-1}{}^i - 2F_n{}^i}{(\Delta r)^2} \quad (21)$$

we get the recurrence relation for the $F_n{}^i$'s

$$F_{n+1}{}^i = k_iF_n{}^i - F_{n-1}{}^i - I_n{}^i \quad (22)$$

where

$$k_i = 2 + \frac{b_i(\Delta r)^2}{D_i}$$

$$b_i = \Sigma_{a_i} + \frac{1}{\omega_2 l^i{}_i\left(\frac{1}{\xi\sigma_s}\right)_{i_{avg}}}$$

$$I_n{}^i = \frac{(\Delta r)^2}{D_i}\left[\left(1+\frac{\omega_1}{\omega_2}\right)_i Q_n{}^{(i-1)\prime} + S_n{}^i\right]$$

Equation 22 is then solved numerically subject to the conditions $F = 0$ at $r = 0$ and also at some suitable extrapolated end-point outside the assembly. In addition, the flux and current must be made continuous at boundaries between different regions in the assembly. We have found it convenient to choose Δr so that the boundaries occur at space points. At such a boundary point, $r_b = N$, we must have‡ $F_N = F_N'$ so that the neutron flux is continuous. For the neutron current to be continuous, the expressions for the currents in the right and left hand media must be equated. The diffusion expression for the neutron current is

$$J = -D\frac{d(NV)}{dr} = -\frac{D}{r^2}\left(-F + r\frac{dF}{dr}\right)$$

The derivative of F with respect to r is not given directly by the difference equation of Eq. 22 so it is necessary to obtain an expression for the derivative in terms of differences. To do this, the cardinal principle has been adopted that the substitution of difference equations for differential equations should not interfere with the conservation of neutrons. This provides an excellent check on the numerical accuracy of the calculations. If the conservation law holds rigorously, any failure to obtain a neutron balance is due to errors in calculation rather than errors introduced by using difference equations instead of differential equations. The expression that must be used for the neutron current is then determined uniquely by the expression that is used for the volume integral of a function.

The integration formula adopted as the starting point is

$$\int_{r_1}^{r_2} r\phi\,dr = \sum_{n=r_1}^{r_2}\phi_n r_n\Delta r - \left(\frac{r_2}{2}\phi_{r_2} + \frac{r_1}{2}\phi_{r_1} + \frac{\Delta r}{6}\phi_{r_2} - \frac{\Delta r}{6}\phi_{r_1}\right)\Delta r \quad (23)$$

This formula is rigorous if ϕ is a linear function of r between junction points. If the conservation law is to hold rigorously, we must have, in general

$$\int_{r_1}^{r_2} r\left(r\nabla^2\frac{\psi}{r}\right)dr = \left(-\psi + r\frac{d\psi}{dr}\right)\Bigg|_{r_1}^{r_2} \quad (24)$$

Using Eq. 21 together with Eqs. 23 and 24, we find by a straightforward calculation that we must have

$$\left(-\psi + r\frac{d\psi}{dr}\right)\Bigg|_{r_2} \to \frac{r_2}{2\Delta r}\left(\psi_{(r_2+\Delta r)} - \psi_{(r_2-\Delta r)}\right) - \frac{2}{3}\psi_{(r_2)} - \frac{1}{6}\left(\psi_{(r_2+\Delta r)} + \psi_{(r_2-\Delta r)}\right) \quad (25)$$

Hence, the neutron current, J, at radius $r = N$ must be represented by

$$r^2J = -D\left[\frac{r_N}{2\Delta r}\left(F_{N+\Delta r} - F_{N-\Delta r}\right) - \frac{2}{3}F_N - \frac{1}{6}\left(F_{N+\Delta r} + F_{N-\Delta r}\right)\right] \quad (26)$$

The quantity on the right must, therefore, be made continuous at boundaries between regions, and its value at the outside edge of the assembly represents the number of neutrons escaping. The fact that the current expression involves a flux value beyond the boundary (e.g., the current flowing from left to right at the radius N involves $F_{N+\Delta r}$§ in terms of the left hand medium solution and involves $F_{N-\Delta r}'$ in terms of the right hand medium solution) causes no practical difficulty since these flux values can be defined math-

* The fundamental reason that calculations of the neutron flux give information on the shape of the adjoint function is the fact that if the correct adjoint function is used in the expression for ν_c, the value obtained for ν_c is independent of the choice of the neutron source shape. Thus, if the fission distributions which result from J linearly independent source functions are known, one can determine the best representation for the adjoint function in terms of J linearly independent functions by the requirement that the same value of ν_c be obtained for each of the J neutron source functions.

† This approximation is equivalent to using an electrical network analogy to the differential equation and was suggested by Gabriel Kron of GE.

‡ In the following, primes will always refer to properties of the region to the right of the boundary. Also, the superscript i referring to a particular energy group will be dropped since all quantities will pertain to the same group.

§ It is sometimes convenient to use different values of Δr in different spatial regions. Hence, in discussing the boundary conditions, we take Δr in the left and $\Delta r'$ in the right hand region.

ematically even though they have no physical significance. The current boundary condition is thus of the form

$$\left[\frac{2}{3}F_N + \frac{1}{6}(F_{N+\Delta r} + F_{N-\Delta r}) - \frac{N}{2\Delta r}(F_{N+\Delta r} - F_{N-\Delta r})\right] = h\left[\frac{2}{3}F_N + \frac{1}{6}(F_{N+\Delta r'} + F_{N-\Delta r'}) - \frac{N}{2\Delta r}(F_{N+\Delta r'} - F_{N-\Delta r'})\right] \quad (27)$$

where $h = D'/D$.

Equation 22 may be solved by starting both a homogeneous ($I_n = 0$) and a particular inhomogeneous solution at $r = 0$ and carrying them independently to the outer boundary. Then the linear combination that goes to zero at the extrapolated end-point is taken as the final solution. This procedure works well only when the group width is large and the absorption is small. Otherwise, serious loss of numerical accuracy is encountered due to the exponential increase of the homogeneous and particular solutions with radius. There are several methods based on the properties of the Wronskian of the original differential equation that avoid this difficulty. R. H. Stark (KAPL) has suggested a procedure that is particularly convenient for solving the equations on fixed decimal automatic computing machines. The method involves introducing new variables α_n and β_n that are related to the homogeneous solution A_n by the equations

$$\alpha_{n+1} = \frac{A_{n+1}}{A_n} \quad \text{and} \quad \beta_n = \sum_{j=0}^{n} A_j \frac{I_j}{A_n}$$

These quantities, which are not sensitive functions of n, may be calculated directly by means of the following recursion relations from Eq. 22

$$\alpha_{n+1} = k - \frac{1}{\alpha_n}, \quad \beta_n = \frac{\beta_{n-1}}{\alpha_n} + I_n \quad (28)$$

The solution for the α's is started with $\alpha_2 = k$. The succeeding α's are bounded and lie between 1 and k ($k \geq 2$). The solution for the β's is started with $\beta_1 = I_1$. Then

$$F_n = (F_{n+1} + \beta_n)/\alpha_{n+1} \quad (29)$$

is solved for the F's from the outside to the center starting with $F_n = 0$ at the extrapolated end-point.

Eqs. 28 have the same mathematical form but different constants at a boundary $n = m$ between two different media. Again letting primes refer to the outermost medium, we have

$$\alpha_{m+1}' = K - \frac{\Delta r'}{s\Delta r}\frac{1}{\alpha_m}$$

$$\alpha_{n+1}' = k' - \frac{1}{\alpha_n'} \text{ for } n > m$$

$$\beta_m' = pI_m + \left(\frac{1}{2} + \frac{\Delta r'}{6r_m}\right)I_m' + \frac{\Delta r'}{s\Delta r}\frac{\beta_{m-1}}{\alpha_m}$$

$$\beta_n' = I_n' + \frac{\beta_{n-1}'}{\alpha_n'} \text{ for } n > m$$

where

$$s = \frac{D'}{D}, \quad p = \frac{1}{2s}\left(\frac{\Delta r'}{\Delta r} - \frac{\Delta r'}{3r_m}\right),$$

$$K = pk + \left(1 + \frac{\Delta r'}{3r_m}\right)\frac{k'}{2} + \left(1 - \frac{1}{s}\right)\frac{2\Delta r'}{3r_m}$$

TABLE 1—Volume Composition

Core	*Reflector*
60.00% moderator	75.00% moderator
10.00% structure	5.00% structure
1.32% fuel	
28.68% void	20.00% void

TABLE 2—Nuclear Properties of Reactor's Components

Component	*Atomic density, N* (10^{-24} *atoms/cm*^{3})	*Microscopic cross sections (barns/atom)* σ_a	σ_f	σ_{tr}	$\xi\sigma_s$
Fuel	0.0478	$101(0.025/E)^{1/2}$	$549(0.025/E)^{1/2}$	$650(0.025/E)^{1/2}$*	0
Structure	0.10	$3(0.025/E)^{1/2}$	0	10	0
Moderator	0.125	0	0	4	0.8

* σ_{tr} is assumed equal to $\sigma_a = \sigma_c + \sigma_f$

TABLE 3—Neutron Balance Sheet*

Range in u of energy group	f_{core}	$f_{reflector}$	*Core fission source*	*Core* Degradation	Absorption	Fission	Escape	*Reflector* Degradation	Absorption	Escape	*Group† source*	*Group‡ removal*
0 – 1	0.25	0.25	0.1250	0.1069	0.0000	0.0000	0.0180	0.0180	0.0000	0.0000	0.1250	0.1249
1 – 2	0.61	0.60	0.4450	0.4581	0.0002	0.0002	0.0936	0.1115	0.0002	0.0000	0.5699	0.5700
2 – 3	0.83	0.74	0.3000	0.6470	0.0007	0.0006	0.1104	0.2207	0.0012	0.0000	0.8696	0.8696
3 – 6	1.05	0.78	0.1300	0.5720	0.0069	0.0054	0.1982	0.3906	0.0281	0.0001	0.9977	0.9977
6 – 9	1.10	0.90	—	0.4447	0.0252	0.0199	0.1021	0.4327	0.0595	0.0006	0.9626	0.9627
9 –12	1.25	0.95	—	0.2956	0.0857	0.0672	0.0633	0.3991	0.0731	0.0238	0.8774	0.8773
12 –15	1.55	1.05	—	0.1158	0.1866	0.1463	−0.0068	0.3063	0.0769	0.0091	0.6947	0.6947
15 –17.5	2.00	1.10	—	0.0201	0.1336	0.1054	−0.0379	0.2066	0.0413	0.0205	0.4221	0.4221
17.5–19.8	3.00	1.25	—	0.0016	0.0473	0.0373	−0.0287	0.1121	0.0275	0.0383	0.2267	0.2268
19.8 (thermal)	—	—	—	—	0.0264	0.0208	−0.0248	—	0.0347	0.0526	0.1137	0.1137
Total, 0–19.8	—	—	1.000	—	0.5126	0.4031	0.4874	—	0.3424	0.1450		

* Normalized to one source neutron in the core. The calculations on the 604 are programmed to keep 5 places to the right of the decimal in the balance quantities. Thus, round-off errors may lead to small discrepancies in this 4-place summary.

† Core fission source and core and reflector degradation from previous group.

‡ Core degradation and absorption and reflector degradation, absorption, and escape.

Equation 29 is unchanged at a boundary since there is just one value of α or β associated with each n. With these modifications, the solution of multigroup problems in the difference equation approximation on computing machines is rapid and simple.

SAMPLE CALCULATION

A sample calculation, illustrating the use of the previously described methods, was performed on an IBM-604 using scheme A and ten energy groups. The reactor that was calculated consisted of a core with a 30-cm radius surrounded by a 20-cm-thick reflector. A spatial interval $\Delta r = 2.0$ cm was used. The composition is shown in Table 1, and cross sections used are shown in Table 2. The U^{235} fission and capture cross sections have been assumed to have a $1/v$ variation with energy and to have the values 549 barns and 101 barns respectively at 0.025 ev.* The scattering cross section of U^{235} is neglected.

The neutron balance sheet, the energy groups, and the f-values are given in Table 3. The uncorrected value of ν_c given by the ratio of the volume integral of the assumed source to the volume integral of the fissions is 2.481. Using the approximation $M_0 = 1 - \frac{1}{2}(r/r_0)^2$, Eq. 19a gives a corrected $\nu_c = 2.484$. Assuming $\nu = 2.5$ for U^{235}, the reactivity of the assembly is $k = 2.5/2.484 = 1.006$.

The assumed and calculated fission source distributions normalized to unit average value are given in Fig. 1. The degradation in the third, sixth, and eighth energy groups as a function of position is given in Fig. 2. The curves illustrate the transition from a cosine-like core distribution with neutrons escaping to the reflector at high energies to the exponential-type core distribution at low energies with neutrons being returned by the reflector. Figure 3 shows the average core and reflector degradation as a function of lethargy. The points are the values at the top and bottom of the energy groups. The straight lines are the average values given by fq^0. It can be seen that the choice of f-values is, in general, satisfactory. Figure 4 shows the lethargy distribution of the spatial average of the core fissions normalized to one source neutron in the core.

* These thermal values were taken from AECU-2040 (1952).

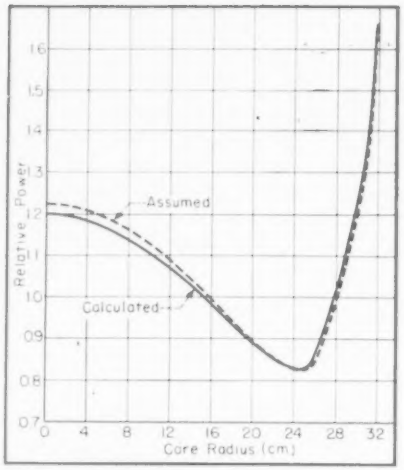


FIG. 1. Core power distribution

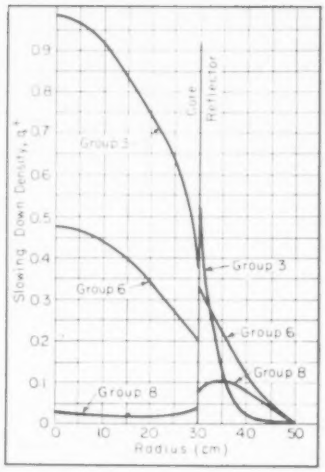


FIG. 2. Slowing down density for three energy groups

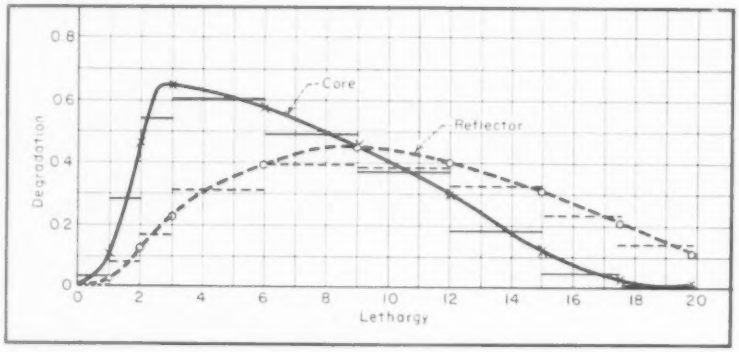


FIG. 3. Space averaged degradation in core and reflector vs lethargy

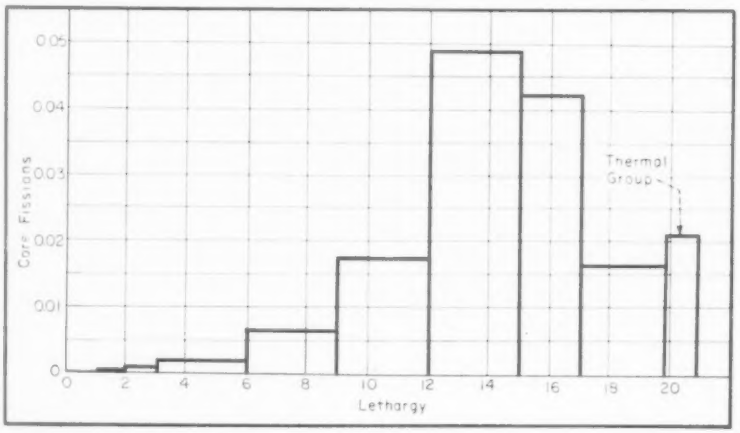
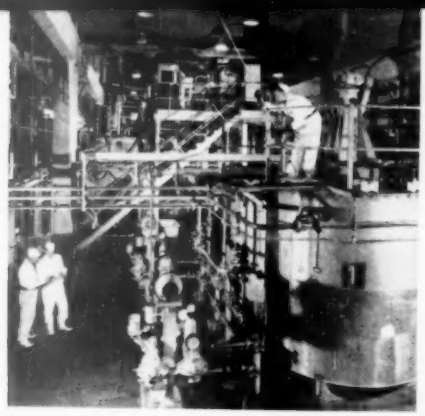
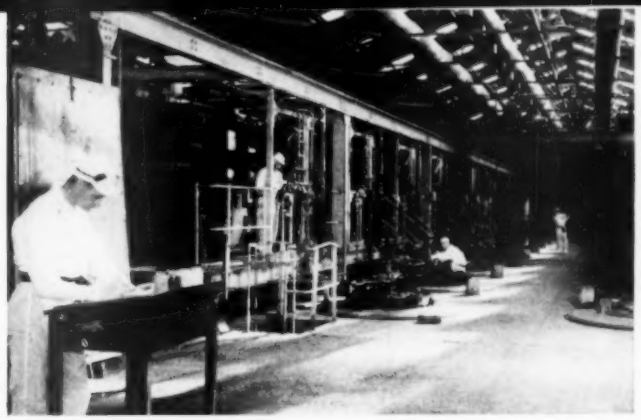


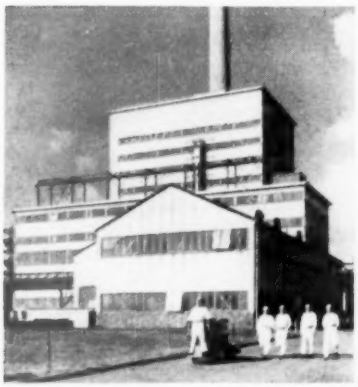
FIG. 4. Core fissions vs lethargy (per core source neutron)

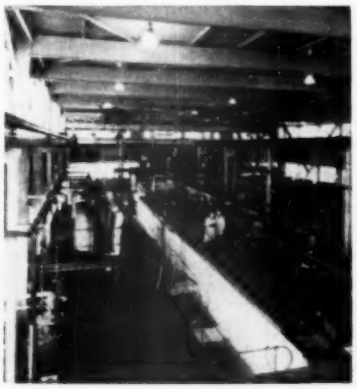
We wish to express our appreciation to P. L. Hofmann and D. Neeb for providing compilations of detailed formulas for all steps in a multigroup calculation, to I. Bornstein for encouraging the use of IBM machines in solving the equations, and to members of the staff of Oak Ridge National Laboratory for interesting discussions of multigroup methods. We are also grateful to the group of F. J. Magginniss (GE) and later the group of R. H. Stark (KAPL) for developing and carrying out machine calculational methods.



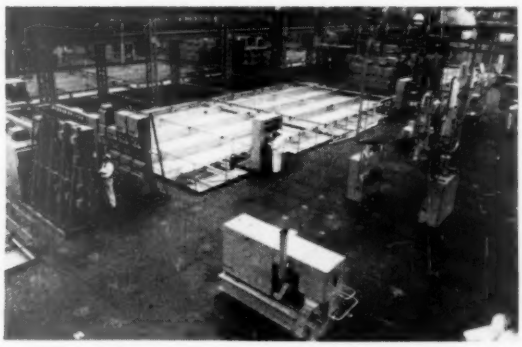


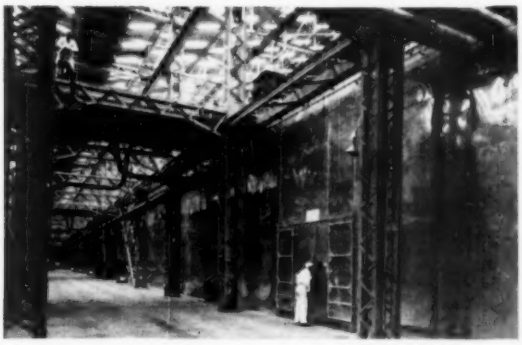
FROM URANIUM ORE . . . At Springfields, ore is powdered and dissolved in acid (left). Tetrafluoride, obtained in "Dryway" process (right), is converted to metal for pile irradiation at Windscale and to hexafluoride for Capenhurst gaseous-diffusion plant





TO PLUTONIUM . . . At Windscale, separation plant (left) recovers plutonium produced in two production piles (buildings in background in right photo); purification plant (center) purifies uranium for conversion of Springfields to UF_6 for Capenhurst





AND TO URANIUM-235 . . . At Capenhurst, uranium hexafluoride is put through gaseous-diffusion cells for U^{235} separation. Process is controlled from deck above cells (left); corridors run between rows of cells (right)

How Britain Produces Fissionable Materials

THREE INSTALLATIONS constructed since 1947 produce Britain's ffssionable materials. Uranium ore is processed at Springfields, plutonium is produced in the reactors at Windscale, and uranium-235 is separated out at the Capenhurst gaseous-diffusion plant.

These pictures are the first released of British atomic energy production facilities.

While gamma rays can provide more uniform distribution of energy in irradiated material, the efficiency of utilization is much lower than for beta particles. Related problems of ionization distribution and bactericidal effects are discussed for both types of radiation in this article

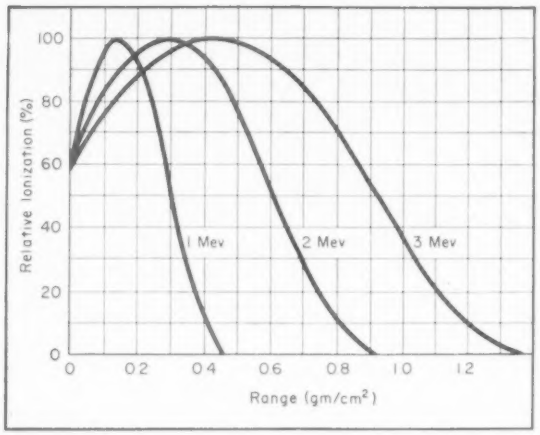


FIG. 1. Distribution of ionization in depth of aluminum produced by cathode rays of different energies (2)

RADIATION STERILIZATION VI

Relative Merits of Cathode Rays and Gamma Radiations

By SAMUEL A. GOLDBLITH and BERNARD E. PROCTOR
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Cambridge, Massachusetts

TWO PROBLEMS are basic to the choice of type of ionizing radiation for sterilization purposes. One relates to the source (machine vs. isotope) of radiation and to the many factors involved in obtaining the desired radiation in sufficient quantity. The other relates to the utilization of available types of radiation insofar as the product to be sterilized is concerned.

Only beta particles (cathode rays) and gamma rays (X-rays) may be considered for sterilization purposes. The reasons for this have been discussed previously (*1*). Other articles in this series have considered the sources of ionizing radiations; this article will discuss utilization aspects when beams of these radiations are of sufficient intensity for sterilization purposes.

Absorption Characteristics

To evaluate intelligently the various factors that must be considered in any study of this nature, one of the first points to be examined is that of absorption of ionizing radiation in matter. There is a radical difference in this respect between beta particles on the one hand and gamma rays on the other.

Energy distribution. With cathode rays or beta particles, the distribution of energy within an absorber is nonuniform. This is illustrated by Fig. 1, which presents curves showing the distribution of ionization in depth of aluminum produced by high-energy cathode rays (*2*). Figure 2 represents a single ionization-in-depth curve for 3-Mev cathode rays, drawn in detail (*3*). It will be observed that the relative ionization varies from 60% at the

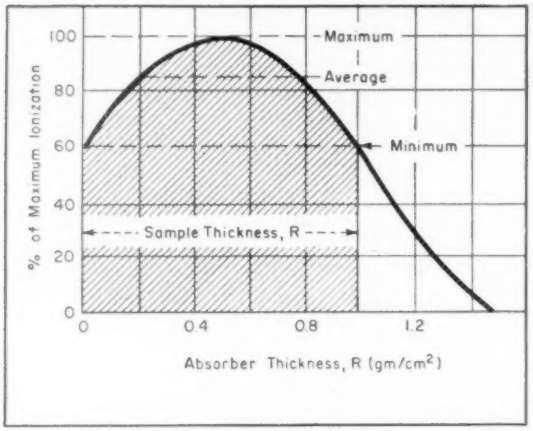

FIG. 2. Ionization in depth for maximum, minimum, and average doses of cathode rays (3)

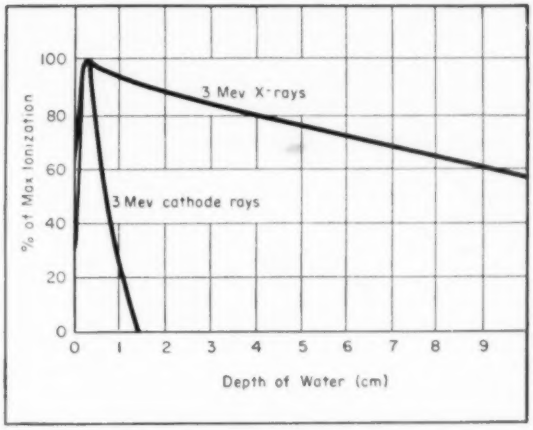

FIG. 3. Distribution of ionization in water by 3-Mev X-rays and cathode rays (4)

surface of the absorber to 100% at one-third of the maximum range of the particular energy electrons. The ionization then falls off rapidly to zero at the maximum range for the particular energy level. Therefore, with a sample having a thickness of 0.55 gm/cm², the variation in dose of 3-Mev cathode rays received from top to bottom of the absorber would be 40%.

The significance of such nonuniform or nonhomogeneous distribution of dose, insofar as the relative bactericidal effect is concerned, will be discussed in some detail later. At the moment, it is important to realize that for specifying power requirements on the basis of a chosen sterilization dose, the experimentally determined dose at the point of minimum ionization level in the absorber should be used in calculations.

Physical and chemical dosimeters give data for dose levels that represent arithmetical average doses. In the case of cathode rays, the average dose may differ considerably from the minimum or entrance dose. For example, with 3-Mev electrons and a sample depth of 0.55 gm/cm², the relative dose at the entrance is the minimum dose and is 60% of the maximum dose, which occurs at the bottom of the sample. The average dose is three-fourths of the exit (in this case, maximum) dose.

Efficiency of utilization. Minimum, average, and maximum doses are as important for gamma rays as for beta rays, if experimental conditions are such that variations in dose of as large a magnitude are found with the gamma radiation. In general, however, variations in distribution of ionization density with gamma or X-rays are not of the same order of magnitude as with beta rays, for equal depths of absorber. Moreover, the absorption of gamma rays (or X-rays) in matter is entirely different (*4*) from that of beta rays (Fig. 3). Absorption of gamma rays is exponential, and the maximum range is not a finite number. Therefore, it appears at first glance that with gamma- or X-rays it is difficult to attain the same degree of efficiency of energy utilization as can be attained with beta rays.

It has been previously pointed out (*5*) that with cathode rays one might expect an efficiency of utilization of energy coming out of the tube of the order of magnitude of 75% if the absorber is irradiated from two sides. With X- or gamma radiation, however, the efficiency of utilization by the absorber is much less. Vitro Corp. (*6*) has calculated that efficiencies of utilization of gamma radiation are much lower: about 14.7% for solid-cylinder sources and 31.8% for hollow-cylinder sources. Calculations made by us, using Vitro data (*6, 7*), indicate an efficiency of utilization as low as 4% for plane sources.

From the point of view of the processor who might use a radiation source for food sterilization, the lower efficiency of utilization of energy from gamma rays means increased source activity requirements if a radioisotope emitting gamma rays is chosen as a radiation source. Furthermore, with gamma rays, as with beta particles, it is necessary from a microbiological standpoint to know the relative importance of any variation in the distribution of ionization density through a product.

Quantitative study of distribution. Although the supplier may be able to provide a sufficiently large isotope source to take care of the increased requirements made necessary by the lower efficiency of utilization, it is still of importance to have quantitative knowledge of the distribution of gamma radiation within a product, in order to be able to specify the minimum dose level within the absorber and thereby to calculate the throughput of product.

Some rough preliminary calculations have been made of the variation in distribution of ionization density within absorbers in the case of conventional packers' cans, which might be bombarded by gamma radiations emitted from two differently shaped geometrical sources. The results of these calculations are shown in Figs. 4 and 5. To obtain the curves presented in these figures, several assumptions have been made, but in general it is felt that these curves approximate the true picture fairly closely when such sources are actually used (*1*).

With isotopic sources of hollow cylindrical shape, it can be seen from Fig. 4 that there should be little variation in dose distribution in an orange-juice can irradiated within a 4-inch source. With a No. 1 picnic can, the variation is only 10% (from 50 to

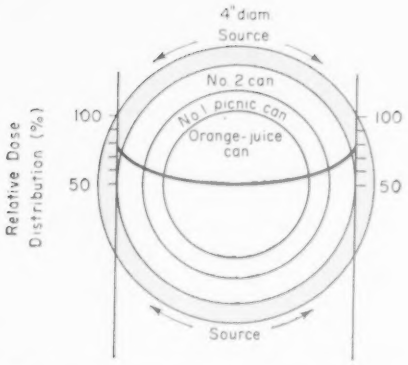

FIG. 4. Variation in distribution of ionization density within absorbers irradiated by an isotopic source of hollow cylindrical shape

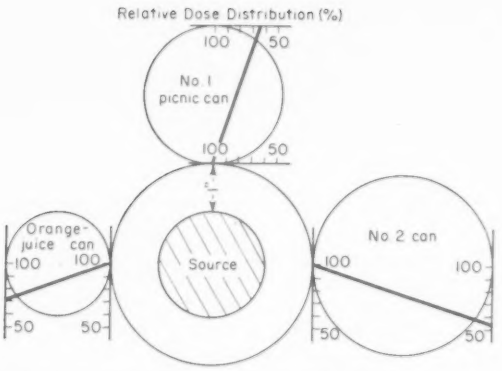

FIG. 5. Variation in distribution of ionization density within absorbers irradiated by an isotopic source of solid-rod shape

60%); with a No. 2 can it is more, namely, 20%.

With isotopic sources, each consisting of a single, solid, rod (Fig. 5) the variation in distribution of ionization density within an absorber is considerably greater, amounting to 28% for an orange-juice can, 36% for a No. 1 picnic can, and 46% for a No. 2 can. In the case of the No. 2 can, the variation approaches that occurring when an absorber 1 cm thick is irradiated with 3-Mev cathode rays.

In all probability, however, multiple-tube or multiple-rod sources would be used, which should improve distribution of ionization density within the absorber and efficiency of utilization of gamma-ray energy. Therefore, these calculations present perhaps the worst possible conditions of energy distribution.

Although research in the field of radiation therapy has made the literature replete with experimental data on energy absorption by materials exposed to point sources of radiation, there is a singular lack of experimental data on absorption by materials exposed to rod-shaped and hollow cylindrical sources. Such studies are of fundamental importance in investigating isotopic sources for sterilization and should be undertaken as soon as isotopes of the spectra available for sterilization purposes are obtainable in sufficient quantity.

Nonuniform Ionization

Nonuniform ionization or nonhomogeneous distribution of radiation intensity may be of importance from two aspects, insofar as the product is concerned. It will result in certain segments of the absorber receiving an overdose and may have an effect on the relative bactericidal efficiency of the particular type of radiation used. The overdose that may occur has been described previously (*8*). Mathematical equations have been derived that indicate the effect of nonuniform ionization on the proportion of organisms surviving a given dose (*8*). These equations demonstrate that when the ratio of entrance dose, D_e (dose at the surface of a sample), to the mean lethal dose, D_0, for a given bacterial species becomes large, the bactericidal effect of nonuniform ionization (e.g., a variation in ionization in depth of from 60 to 100% in the case of cathode rays) becomes very great. For example, in the case of a spore-forming species of bacteria, the entrance dose might be 2×10^6 rep and the mean lethal dose for sterilization purposes 50,000 rep, which means that the ratio would be 40:1.

This relation between the ratio of entrance dose to mean lethal dose and the effect of nonuniform ionization is illustrated in Table 1. The data show that if a culture of bacteria is exposed to the same average dose of high-energy cathode rays, pronounced differences occur in the relative bactericidal efficiencies of homogeneous and nonhomogeneous beams when doses greater than four times the mean lethal dose are used.

These mathematical evaluations have been confirmed experimentally by exposing to 3-Mev cathode rays cultures of *E. coli* of thicknesses of 0.1 and 0.51 cm, that is, to thicknesses sufficient to achieve, respectively, uniform ionization and a distribution of ionization density of 40% through the sample. Confirmation of the effects of the uniform or homogeneous distribution of ionization density was further obtained by exposure of ampoules of the same culture to gamma rays from a cobalt-60 source.

From the preceding discussion it is apparent that lack of uniformity in the distribution of ionization density within an absorber decreases the relative bactericidal effectiveness of the radiation. In the case of 3-Mev cathode rays, the ratio of mean lethal dose for uniform ionization to mean lethal dose for nonuniform ionization is experimentally about 6:10. For example, with uniform ionization an

TABLE 1—Theoretical Comparison of Bacteriological Effect for Samples Given the Same Average Dose*

D_e/D_0	n (nonuniform) / n (uniform)	D_0 (nonuniform) / D_0 (uniform)
0.25	1.00	1.00
0.5	1.00	1.00
0.75	1.01	1.01
1	1.02	1.01
1.5	1.04	1.02
2	1.07	1.03
3	1.17	1.04
4	1.35	1.06
6	1.81	1.08
8	2.69	1.09
10	4.21	1.12
20	58.9	1.18
40	2280	1.23

* n = survivors; D_e = entrance dose; D_0 = mean lethal dose.

absorption of approximately 6,000 rep is required to effect a 63% destruction of bacteria, whereas with nonuniform ionization an absorption of 10,000 rep is required.

From a practical standpoint this is of importance in food processing, because it is unlikely that in such processing very thin samples will be available for most operations. On the contrary, it is expected that the radiations used for sterilization purposes will have to penetrate very thick samples, and if nonuniform distribution of ionization density occurs, the relative bactericidal effectiveness of that particular radiation will be less. Therefore, it behooves anyone interested in utilizing either beta or gamma radiation to achieve as uniform as possible distribution of the beam within the absorber.

Obviously, for samples of equal thickness, the distribution of gamma radiation is better than that of beta radiation. However, because gamma radiation is recommended for sterilization of conventional packers' containers (rather than for containers of 1 to 2 cm thickness), and because with containers of such thicknesses there is considerable variation in the distribution of ionization density, it is of the utmost importance to improve the distribution of ionization from solid-rod and plane sources. This may be achieved in part by using multiple-rod instead of single-rod sources. However, experimental confirmation of this assumption is still needed.

Next month in NUCLEONICS:

Scintillation Counting Today

An up-to-date report of practice and theory, including a detailed review of the Scintillation Counter Symposium plus many of the major papers presented in full

Bactericidal Effects

From the standpoint of utilization of radiation, it is also of importance to know the relative bactericidal effects of *equal* doses of beta and gamma radiations. Reports are notably lacking of intensive comparative studies on the relative bactericidal efficiencies of beta and gamma radiations under strictly comparable conditions of irradiation dosimetry and bacteriological techniques. Attempts to correlate the scattered information in the literature have revealed the inconsistencies of available published data as well as problems that one might expect to encounter.

Recently, studies have been initiated in the Food Technology Laboratories at Massachusetts Institute of Technology under sponsorship of the U. S. Atomic Energy Commission, to compare the effects of beta and gamma radiation from a microbiological point of view. These studies were made under strictly comparable and controlled conditions of dosimetry and bacteriology.

Some of the data obtained for *E. coli* and *B. thermoacidurans* are summarized in Table 2. They indicate that there is no essential difference in the bactericidal effects of beta and gamma radiation, provided that in the case of beta rays the sample is thin enough for uniform distribution of ionization density to be attained. When, for instance, a sample of 0.51 gm/cm² thickness is chosen for irradiation by 3-Mev cathode rays, beta radiation does not appear as effective as gamma radiation because of the nonuniform ionization. Further investigation is underway in these laboratories on other spore-forming organisms.

* * *

TABLE 2—Comparison of Effects on Bacteria of Gamma Rays from Cobalt-60, and X-Rays and Cathode Rays from a Van de Graaff Accelerator

Organism	*Source*	*Container*	*Sample volume*	*Number of samples*	*Mean D_o* ($rep_{93} \times 10^{-3}$)
E. coli	Van de Graaff cathode rays	Petri dishes (large)	5 ml (0.081-cm depth)	54	5.23
E. coli	" " "	Petri dishes (small)	2 ml (0.10-cm depth)	52	6.21
E. coli	" " "	Petri dishes (small)	10 ml (0.51-cm depth)	52	10.3
E. coli	" " "	Ampoules	3 ml	61	7.48
E. coli	Van de Graaff X-rays	Ampoules	3 ml	108	5.64
E. coli	Cobalt-60	Ampoules	3 ml	196	6.45
B. thermoacidurans	Van de Graaff cathode rays	Ampoules	3 ml	118	50.1
B. thermoacidurans	Cobalt-60	Ampoules	3 ml	157	45.6

The authors are grateful to Dr. Sol Davison, Department of Food Technology, MIT, for the drawings shown in Figs. 4 and 5 and for his contributions on the theoretical and experimental comparisons presented in Tables 1 and 2.

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Logarithmic D-C Ratemeters for Scintillation Counters

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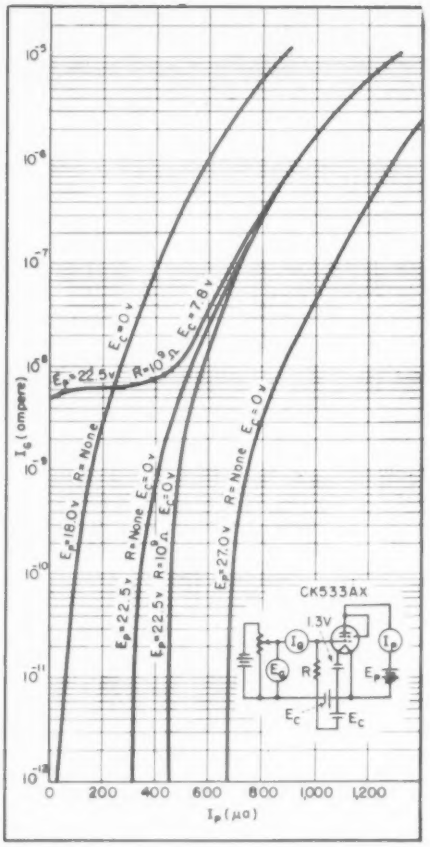


FIG. 1. Grid current vs plate current characteristic of CK533AX

Five or more decades can be covered without range switching by ratemeter circuits that utilize the logarithmic response characteristic of a subminiature triode. Advantage can be taken of triode photomultiplier characteristics for simple adjustment of mass-produced instruments to a standard calibration

WHILE PULSE TECHNIQUES using amplifiers and shaping circuits are stable under normal conditions, the multiplicity of batteries and circuits needed for portable units usually involve major servicing problems. By a proper selection of circuits and tubes it is possible to create highly sensitive, simple ratemeter equipment that will not only perform as well as pulse circuits for portable instruments but will have added features such as extended dynamic range, high current output in the reading circuit, stability over long time periods, and simplified circuitry with increased instrument reliability.

This article describes the development of such circuits for portable equipment and other applications where the incoming rate of data is adequate to supply good statistics.

Logarithmic Ratemeter Circuit

Examination of the characteristics of a number of filamentary subminiature tubes indicates that the Raytheon CK533AX has a marked advantage in the logarithmic response characteristic obtainable from the relationship of grid current versus output plate current. Figure 1 illustrates the possibility of securing at least six decades of information without switching or other techniques, information that is practically logarithmic in response and, as shown later, permits reproducible calibration characteristics without elaborate circuits. In view of the voltage swing when the grid is permitted to establish operating characteristics, the grid current versus grid voltage characteristics are of interest. This information is shown in Fig. 2.

As shown in Fig. 1, for a plate supply of 22½ volts, there is a difference be-

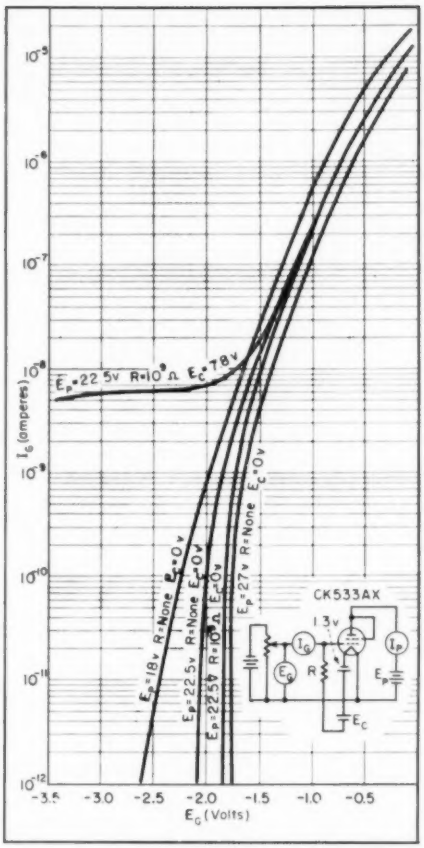


FIG. 2. Grid current vs grid voltage characteristic of CK533AX

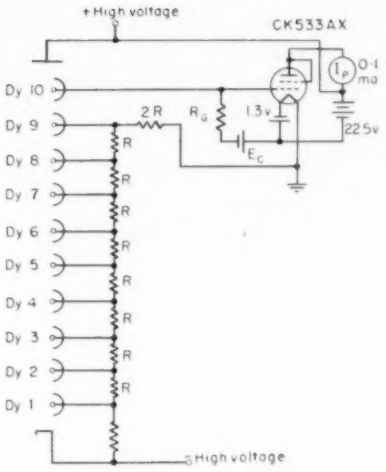


FIG. 3. CK533AX with photomultiplier

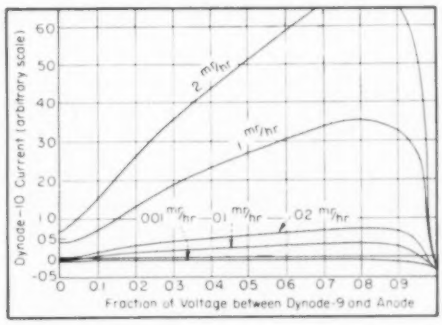


FIG. 4. Current variation in dynode-10 as its potential varies

tween data for an open grid and for one with 10^9 ohms; with zero biasing voltage under the 10^9 ohm resistor, the curve shifts to the right. It is possible by applying biasing voltage, E_c, to return the curve not only to the condition of open grid but to swing the curve in continuation of the straight-line characteristic from 10^{-9} down to 10^{-12} amperes. It is also possible to distort the curve with excessive bias voltage, as shown where E_c is brought to 7.8 volts. Note that the response curves are practically parallel with a change of plate supply voltage from 18 to 27 volts. Figure 2 reveals that the grid swing does not exceed $2\frac{1}{2}$ volts with a $22\frac{1}{2}$-volt plate supply; at 18 volts the dynamic range is increased.

The direction of electron flow is from the cathode of the CK533AX triode to the grid, and therefore the photomultiplier structure is connected as shown in Fig. 3, where the last dynode is connected directly to the grid of the readout tube (CK533AX). It will therefore assume a potential determined by the grid voltage versus grid current characteristics of the connected tube. Figure 4 illustrates the general characteristics of the DuMont structure as

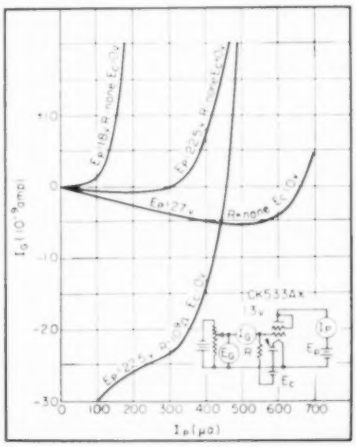


FIG. 5. Grid current vs plate current characteristic of CK533AX for low I_p

one measures the dynode-10 current, keeping the potential between dynode-9 and anode constant, and increasing the voltage between dynode-9 and dynode-10 up to anode potential. An activated sodium iodide crystal $1\frac{1}{8}$-in. diameter by $1\frac{1}{2}$-in. high was used with a radiation source as a light source.

A potential of at least 10% of the voltage between dynode-9 and anode must exist between anode and dynode-10 in order that a negative characteristic does not result. The data in Fig. 4 reveal a negative current for 0.01 mr/hr. This is the net current resulting from leakages within the multiplier tube and is compensated by the biasing voltage applied underneath the 10^9-ohm resistor.

Figure 5 illustrates, with linear-scale ordinate and abscissa, the low-end characteristic of grid current versus plate current for a CK533AX in triode connection. It can be seen that from open grid to 10^9 ohms of external grid

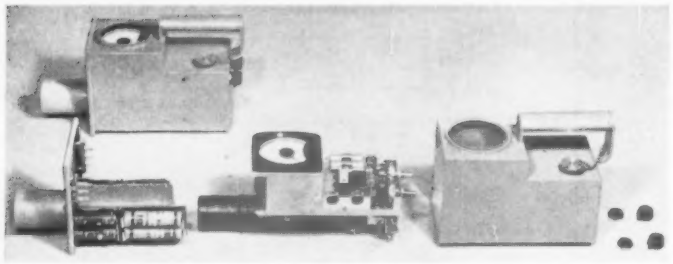


FIG. 6. Assembled and disassembled Scintameter

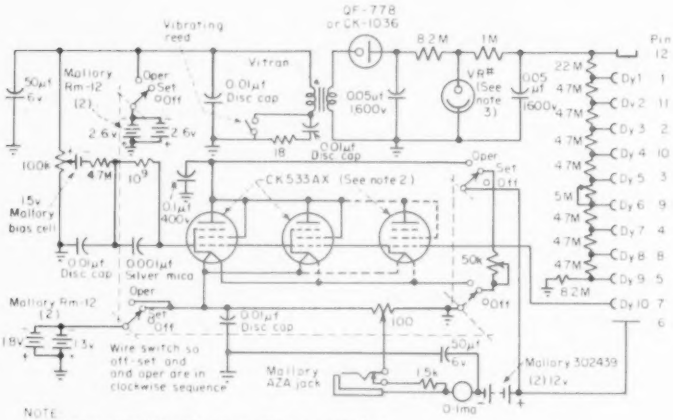


FIG. 7. Circuit of Scintameter

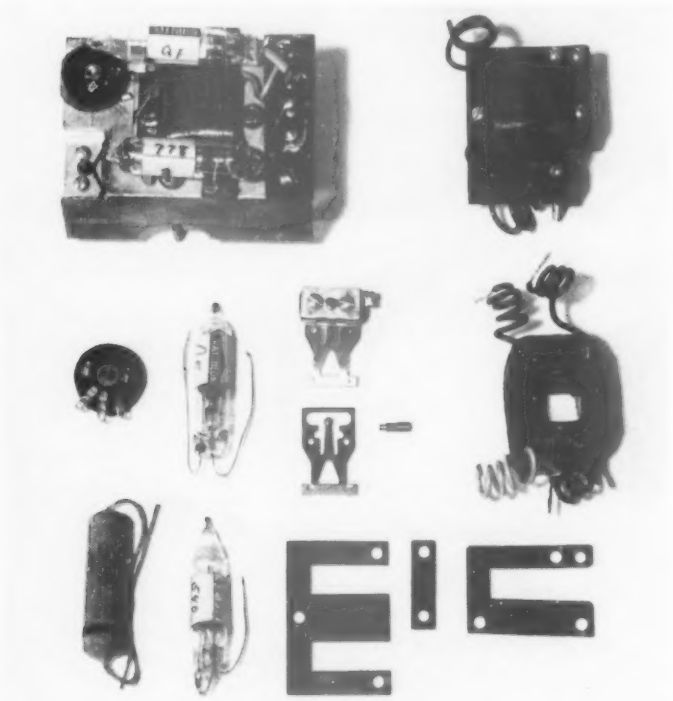


FIG. 8. Components and assembled Vitran unit used in Scintameter

resistance, the curve characteristic is not markedly altered, except that the portion of the curve that crosses zero current is a relatively straight-line continuation from the higher I_g characteristic and therefore provides an extension of the dynamic range. The adjustment of the biasing voltage, E_c, places this portion of the characteristic in a useful region as well. However, by varying the bias voltage underneath this resistor it is possible to shift the "float point" for correction of the low-end characteristics. In the operating range used, the dynamic grid resistance will vary, based on the data shown, from a very high value down to approximately 1 Meg. It is difficult to evaluate the functioning within the multiplier tube as a result of this circuit connection; however it is believed that the multiplication of dynode-10 is always more than unity.

Portable Instruments

Using the basic circuit of Fig. 3, a survey meter for gamma-ray detection has been designed. Called the "Scintameter," it is shown in Figs. 6 and 7. Two CK533AX tubes are used in parallel—and it seems as if the input current divides reasonably well. High voltage is derived from a Vitran* shown in Fig. 8. This unit provides regulated filtered voltage to a maximum of 1,600 volts at 20 to 30 μa, with a ripple of less than ½ volt. Efficiencies are well over 30%, using an input of 2.6 volts at 50 ma.

The detector element of the Scintameter is a 1⅛-in. diameter by 1½-in. high NaI(Tl) crystal.† A typical Scintameter calibration is illustrated in Fig. 9. On the basis of constructing 30 complete survey-meter units and the checking of more than 50 multiplier tubes, it is possible to calibrate more than 80% of the tubes in all the circuits so that the typical characteristics are maintained within a few per cent.

All instruments now fabricated at the Health and Safety Laboratory, New York Operations Office, are designed for direct coupling to an Esterline-Angus recorder with 0 to 1.0-ma range.

Where a lesser dynamic range is desired—four instead of five decades—

* H. D. LeVine, H. DiGiovanni. An inexpensive wide-range gamma-ray survery meter, NYO-1538 (1953).

† Harshaw Chemical Co., Cleveland, Ohio.

three CK533AX tubes may be used in parallel to provide a range of readings from 0.005 mr/hr to 10 mr/hr with 1⅛-in. diameter by 1½-in. high NaI-(Tl) detector at a useful plate current of 1 ma at maximum reading. This modification of the Scintameter has been designed for prospecting use.

Circuit calibration. The three mechanisms for adjusting a survey meter to a standard of characteristics are:

1. The adjustment of the bias control to alter the position of float of the grid and set the lower portion of the characteristics.
2. Adjustment of the gain of the multiplier tube to adjust the characteristic slope.
3. Adjustment of the plate supply voltage of the CK533AX to slide the characteristics over to the desired position.

Generally the first two parameters are used as controls and the third only when the curve cannot be brought into a match with the standard. Practically all CK533AX can be so adjusted and only those which we suspect are gassy or have poorly processed cathodes, are not useful.

Altering the size of the crystal detector will alter the sensitivity of the circuit. A group of instruments with a calibration of Fig. 10 have been constructed for high levels of radiation, 0.001 r/hr to 10 r/hr. The phosphor used here is a plastic phosphor,‡ 1 in. in diameter by ½ in. high.

A gamma-ray drill hole logging probe has been designed using this circuit and is used with the portable, electrically driven reel illustrated in Fig. 11. This type of probe is especially useful since the dynamic range from no activity to ore is in the order of five decades of information; a linear detecting circuit will not cover this spectrum of sensitivity conveniently.

To expand the reading capability of this device, a transistor circuit has been successfully used and displays good stability and functioning characteristics. It is possible to expand the full-scale reading from two to three to four decades by a simple switching arrangement.

This type of circuit should also be useful for neutron flux measurements and other conditions where the wide dynamic range is needed. Because the low output impedance of the reading circuit may be coupled into a transistor, it is also feasible to use long transmission lines where the pulse rise time and decay will be a function of the line characteristic.

‡ Pilot Chemical Company "Scintillation B."

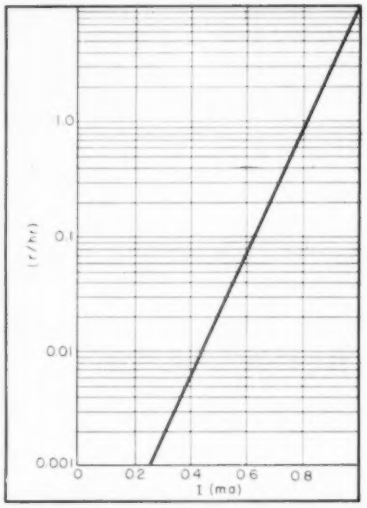


FIG. 9. Representative calibration curve for Scintameter with NaI(Tl) phosphor

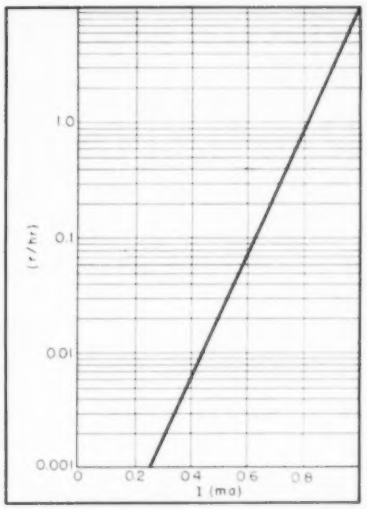


FIG. 10. Calibration curve for Scintameter with plastic phosphor

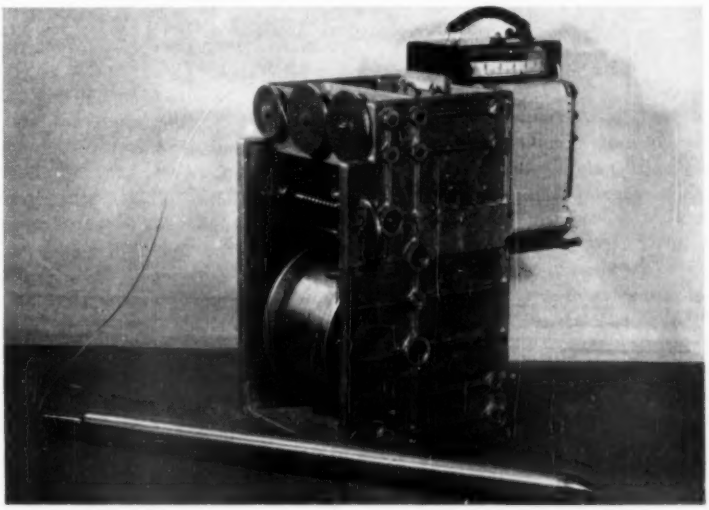


FIG. 11. Portable drill-hole probe with reel, used with Scintameter circuit

Summary

The circuit discussed displays excellent stability characteristics, has a reproducible calibration and involves simple circuit techniques. Because of the resistance values used in the grid circuits the equipment will be affected by humidity; since present day techniques involve sealed apparatus, this is not a major problem with use of hermetic sealing methods and desiccants. The Scintameter operates without trouble at 10,000-ft elevations. The use of mercury cells is advisable since their voltage characteristic is practically flat over their normal life time.

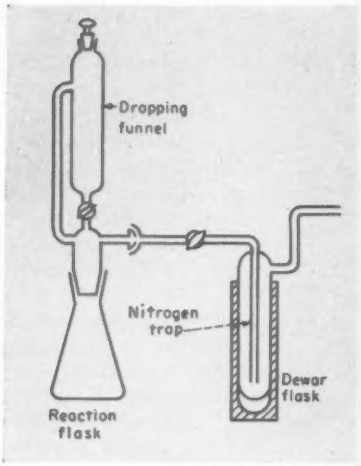
It should be noted incidentally that these same techniques are applicable to Geiger tubes as well. Also suggested is use of the logarithmic circuit for pulse-compression techniques for a wide dynamic range of pulse heights.§

§ Private communication, L. Wouters.

* * *

Ole Pedersen, R. Graveson, M. Cassidy, H. Sadowski, A. Yoli and S. Watnick of this laboratory have materially aided in this work and the author gratefully acknowledges their assistance. The bulk of the data has been gathered by Mr. Pedersen.

How to Prepare and Use Radioactive Soils



Detergency tests have long been plagued by the inability to measure unremoved soil. The authors discuss three types of labeled soils that can easily be measured

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Easton, Pennsylvania
and
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ONE DIFFICULTY in testing detergents or soil-removal qualities in the laboratory has been the problem of detecting soiling agents on the cleaned substrates. A washing performance can be anything from "good" to "poor" depending on the effectiveness with which the last traces of ingrained soil are removed (*1, 2*). To meet this problem, the authors considered radioactive labeling of carbon constituents of a synthetic multi-component soil described by Sanders and Lambert (*3*). In the ensuing cooperative work, three types of labeled soils were jointly developed at our laboratories.

Selection of Labels

The synthetic soil (*3*) consisted of 15 components, including humus, fatty and colloidal substances, and small amounts of iron oxide pigment and carbon black. Although carbon black has been the most predominant soiling ingredient in practically all conventional laboratory wash tests, it was shown that soiling with ultra-fine carbon blacks in large quantities has given unreliable results (*2*). Nevertheless, a certain small amount of medium-fine carbon black in the presence of large quantities of other materials seemed necessary for the formulation of a realistic synthetic soil (*3*). Uniformly C^{14}-labeled carbon black of the desired particle size was therefore selected as the one ingredient that would impart a highly specific, traceable component.

A subsequent search for contaminants that could be used for decontamination experiments in the laboratory led to the admixture of fission products to inactive synthetic soil. It was found that long-lived mixed fission products seem to adsorb primarily onto the colloidal constituents of the soil; thus introducing a nonspecific general label of great utility.

Another attempt was made to introduce an over-all activity into the synthetic soil by neutron irradiation. Although a sample of suitable characteristics was obtained in a single irradiation experiment, it is fairly certain that the attained activation would be greatly influenced by the presence of uncontrollable traces that will vary in different batches of synthetic soil.

Labeled Synthetic Soil

The previously described soil composition (*3*) contained 1.5% Molacco Furnace Black‡ of an intermediate particle size (average diameter: 0.08 μ) corresponding closely to that of airborne carbon (*4*).

Suitability of the C^{14}-labeled carbon black to be incorporated into the synthetic soil was determined beforehand by means of five inactive runs following the procedure described later; three samples were tested for pigment properties (*5*), and one sample was studied in the electron microscope in comparison to several other commercial carbon blacks.

It was found that the characteristics of the experimental carbon black (purity: 99.1–99.9%) were very similar in every respect to the Molacco brand. The nature of the surface of the carbon blacks was determined by measuring the *p*H values of aqueous slurries according to a procedure described by Wiegand (*6*). Molacco Black exhibited a high *p*H in this test (about 9) while the experimental carbon blacks varied from *p*H 2.8 to 4.8. Therefore a heat treatment (*6*) was applied to raise the *p*H above 7.

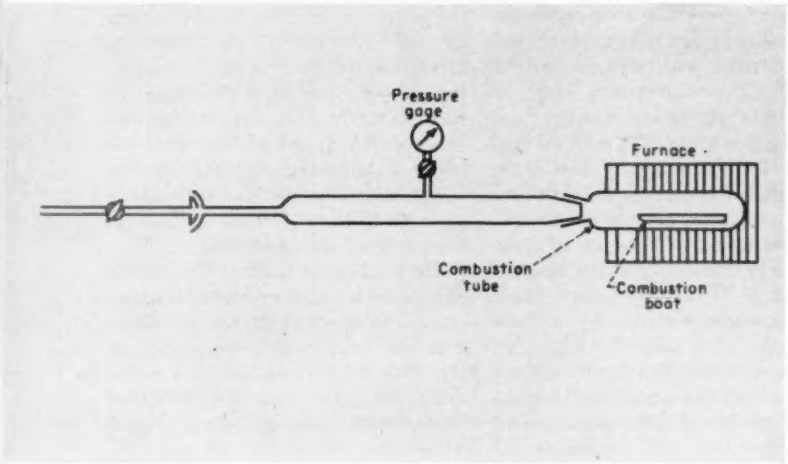
The active and inactive carbon blacks were prepared in the apparatus shown on this page, using a procedure based on the work of Campbell and Brown (*7*). A uniformly labeled product of reduced specific activity was obtained by simultaneous generation of active and inactive CO_2.

In a typical active run, about 0.08 mole of inactive $BaCO_3$ was mixed with 3 mc of C^{14}-labeled $BaCO_3$, placed in the reaction flask, and moistened with distilled water. Lactic acid was dropped slowly from the funnel and the evolving CO_2 was frozen out in the trap cooled with liquid nitrogen.

* PRESENT ADDRESS: Mellon Institute, Pittsburgh 13, Pa.

† PRESENT ADDRESS: Polytechnic Institute of Brooklyn, Brooklyn 2, N. Y.

‡ Manufactured by Columbian Carbon Co. and sold by Binney and Smith Co., New York 17, N. Y.



◀ APPARATUS for preparing radioactive carbon black

After a sufficient amount of gas had condensed, the generation was halted and the trap cautiously permitted to warm; a controlled amount of CO_2 thus entered the combustion tube where it reacted with 5 gm of chemically pure manganese in the quartz boat at a temperature of about 600° C. Since the consumption of CO_2 was indicated by a sharp pressure drop, the reaction could be periodically restarted and safely controlled.

After completion of a run, the contents of the boat were treated with 18% HCl and passed through a fritted glass filter, and retained carbon was washed with distilled water and alcohol and dried. Different and possibly more reliable methods for obtaining labeled carbon blacks of desired characteristics were recently described by Kirshenbaum, Hoffman, and Grosse (*8*), who claimed that the activation could best be accomplished by reacting a commercial black with active CO_2 at about 1,000° C.

The approximate activities of the carbon blacks and of the processed soil were determined by counting each sample in a cupped aluminum planchet 5 mm from a 1.5 mg/cm² mica end-window G-M tube; a sample of $BaCO_3$ of known specific activity served as the reference. The specific activity of the carbon blacks prepared by the CO_2-manganese method varied from 1.4 mc/gm to 4.0 mc/gm; a sample that had attained the highest specific activity was incorporated into a synthetic soil of the usual composition. The procedure outlined previously for the preparation of inactive synthetic soil (*3*) was essentially followed, but the dried soil slurry was ground and sifted in a dry-box to avoid contamination and personnel hazards. A typical batch of C^{14}-labeled soil (identified by No. 2579-142) was prepared by using 0.8 gm carbon black¶ (3 mc) to make 53.2 gm of synthetic soil (60 μc/gm). A very small amount of soil weighing only 0.1 mg, when applied uniformly on a cotton swatch by spreading and rubbing the soil into a 3-in² circular area, gave over 1,000 cpm.

Fission Product, Soil Mixtures

The mixed fission products were received in HNO_3 solutions of high specific activity (approximately 1 mc/ml); they were usually added to the extent of 1 part in 1,000 to a 0.2% synthetic soil slurry in distilled water. The resulting mixture gave a counting rate of about 10,000 cpm per drop, and further dilution was necessary for detergency studies. This was accomplished by adding the mixture in suitable proportions (1 part into 15) to inactive soil slurries that varied in concentration from 1 to 2%.

After decay for six months or longer, the main fission-product activity consists of relatively few isotopic cations and their active daughters (*9*). We found that essentially all active constituents of the aged fission products adsorb uniformly onto the colloidal constituents of the synthetic soil. The same adsorption mechanism has been used successfully by Hatch and Ginell (*10*) for disposing of radioactive cations and fission products; they also showed that the adsorbed materials can be permanently fixed onto montmorillonite and other clay adsorbents by heat treatments at varying temperatures up to 1,000° C. Since the present composition of the synthetic soil would not permit heating to high temperatures without burning or charring the organic components, no heat treatments were applied to the mixtures.

By filtering several soil slurries through Whatman No. 50 filter paper, we ascertained that over 90% of the activity is carried by the particulate matter of the suspensions. Furthermore, by preparing a special series of active slurries and adjusting their *p*H values, we found that only very slight and probably insignificant desorption of fission products takes place under the alkaline conditions usually prevailing in detergency tests.

One sample of mixed fission products used in our studies had been stored less than six months, the other about two years prior to shipment (*11*). Nevertheless, the radiation and decay characteristics of the two samples (determined at the time when the first batch was well over one year old) were found to be very similar; the respective data fell within the experimental error of the determinations. The apparent half-life of the mixed fission products was determined by comparative measurements over a period of nine months; an approximate value of 14 months was determined. This is in fair agreement with published data (*9*) for the decay of total beta power in aged fission products.

The heterogeneous nature of the beta and gamma radiation emitted by mixed fission products makes it impossible to describe in a simple manner the absorption characteristics of the total radiation. However, when synthetic soil labeled with fission products was applied to cotton swatches, the radiation measured by a thin-window G-M tube consisted predominantly of soft and hard beta rays and could be characterized by two apparent half-value thicknesses. The one value appears to be valid in the range from no absorber to 15 mg/cm² Al and the other in the range of thicker absorbers up to 500 mg/cm² Al. Under the

¶ The initial *p*H value of a slurry made with this active carbon black was 2.3; after a short heat treatment for 10 min at 900° C in a N_2-atmosphere, *p*H rose to 7.2.

described conditions a half-value thickness of 25 mg/cm² was found for the first range (soft beta) and a value of 200 mg/cm² for the second (hard beta) range.

Neutron-Irradiated Soil

Some inactive synthetic soil was irradiated in the Brookhaven reactor for 159 hr at a neutron flux of 2.8×10^{12} n/cm²/sec. After irradiation, the soil was permitted to cool for seven days to allow short-lived nuclides to decay almost completely. Under our counting conditions, the approximate specific activity of the irradiated soil was determined to be 0.1 mc/gm. Considering the relative abundance, neutron cross-section, and decay of the constituents in the soil mixture, it has been assumed theoretically that Fe^{59} was the major source of the activity induced into the synthetic soil. It was found that this soil imparted very impermanent activity into most substrates, indicating that all radioactive components present in the mixture can be too easily removed.

Utilization of Labeled Soils

Concurrently with a study of radiological decontamination not yet published, a related investigation was carried out with particular emphasis on the selection of suitable detergents and detergent compositions for removing fission products effectively (*12, 13*). The role of sequestrants in decontamination was also successfully studied (*14*). The use of the C^{14}-labeled soil in realistic cotton detergency testing led to: 1, development of modified soiling procedures; 2, introduction of pre-washing cycles in water alone; and 3, promising experiments with active soils containing simultaneously C^{14}-labeled carbon black and fission products.

In these detergency studies a Schiefer Abrasion Meter was used for soiling the cotton swatches (*3*), but no dry applications of radioactive soils were attempted. A safe procedure consisted of pipetting 0.1-ml portions of a dilute (usually 2%) aqueous soil slurry onto pre-wetted swatches clamped in the abrasion meter; the meter was equipped with a frosted glass disk that was started manually and stopped automatically after 20 revolutions.

By comparing preliminary washing results obtained with several reference detergents and with plain water as a control, we noted that, contrary to the performance in practical washing, hot water removed relatively too much of the soil. This discrepancy could be overcome by rinsing the swatches prior to soiling in a 0.1% CCl_4 solution containing the mixture of the same seven fatty substances used in the synthetic soil (*3*). Another important observation was made in the course of these preliminary tests: The small residual amounts of C^{14}-labeled carbon black measured on swatches after realistic multi-cycle wash tests did not at all correlate with the visual appearance of the cloth. This finding led to the related study of the influence of chromaticity in the evaluation of whiteness described by Hemmendinger and Lambert (*15*).

Finally, experiments were made with a synthetic soil containing both C^{14}-labeled carbon black and mixed fission products at the same time. In the 2% soil slurry (Exp. No. 2909/183) used most extensively, the two respective activities were adjusted so that each contributed about one-half of the counts after initial soiling.

In quantitative soil removal tests, counting was postponed until after the first pre-wash in hot water, since the removal of superficial soil greatly improved the reliability and reproducibility of the tests. The pre-wash liquors containing a definite amount of soil in finely divided state were saved and used later for preparing the detergent solutions for the actual wash; the soil concentration during washing could thereby be increased to a level closer to practice.

The radically different absorption characteristics of the very soft and the relatively hard radiations emitted by C^{14} and mixed fission products respectively, made a complete and quantitative separation possible by counting first without absorber and subsequently with a 15 mg/cm² Al filter. The latter measurement gave an apparent counting rate of the fission products; for obtaining the actual counting rate a correction had to be applied to compensate for absorption of the soft component. Under the prevailing counting conditions and geometry, a multiplying factor of 1.4 rendered the results comparable to those measured without absorber.

From the counting rates of each swatch before and after washing in the detergent solution, the per cent soil removed, SR, can be determined by $SR = 100(c_p - c_w)/c_p$, where c_p and c_w are the net counting rates after the pre-wash and after the wash, respectively. In this manner, removal of carbon black and miscellaneous colloidal contaminants could be followed independently in the course of multi-cycle wash tests. Preliminary findings have been discussed elsewhere (*15*).

The hard component of the radiation emitted by the fission products was also utilized to good advantage for determining trace amounts accumulating on the cloth by re-deposition in a multi-cycle test. Since four clean swatches were always added as ballast to each wash load, this cloth was used for making up a folded specimen of 24 cloth layers. With the absorber in place, the bulk of the specimen greatly increased the sensitivity so that precise re-deposition data could be obtained on swatches that appeared visually perfectly clean.

* * *

The radioactive barium carbonate and mixed fission products used in these studies were supplied by Oak Ridge National Laboratory under authorization from the U. S. Atomic Energy Commission. The neutron-irradiated soil was irradiated at Brookhaven National Laboratory under similar authorization.

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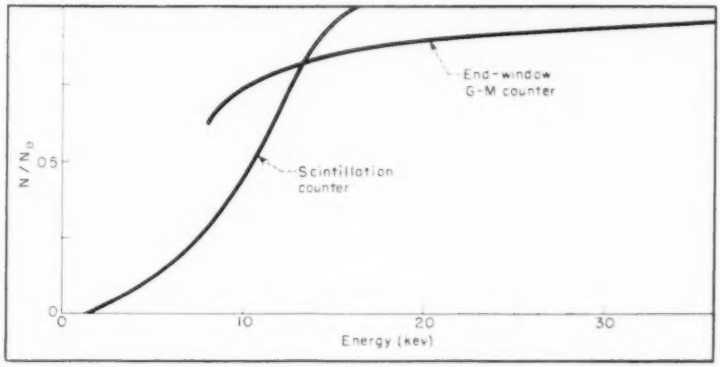

FIG. 6. Typical curves of ratio of number of electrons counted, N, to number impinging on detector, N_0, for anthracene (73); and G-M tube (54)

Experimental Techniques in Beta-Ray Spectroscopy—II

Use of GM and scintillation counters, and electron multipliers as detectors for beta-ray spectrometers poses many problems, as does producing, measuring and controlling the required magnetic field. Techniques used to surmount these problems are considered in this article, the conclusion of a summary of basic data

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In Part I of this article (NU, Dec. '53, p. 28), useful techniques for source preparation, methods of gamma-ray measurement, and design problems associated with the intermediate, evacuated regions of a beta-ray spectrometer were considered. Continuing this compendium of experimental techniques, this concluding section of the article deals with detectors and the magnetic field.

Two types of detectors are generally used in magnetic beta-ray spectrometers. Most common is the Geiger-Müller counter; the second is the scintillation counter.

Geiger-Müller Counter

Theory and operation of G-M counters has been adequately discussed by several authors (*38*).† But use of a G-M counter with a magnetic spectrometer involves some specific problems that are important because most of the counters for this use must be constructed in the laboratory to provide the thin windows needed. For consideration here, these problems are classified as pertaining to counter construction, gaseous filling, thin window, and use of accelerating potentials.

Construction. To make a G-M tube operate requires no special techniques. But to make it operate uniformly over long periods of time (a necessity in spectrometers) requires primarily cleanliness. Extreme care must be taken that all solder joints are left clean; any excess flux must be washed away. Otherwise interaction between the counter gas and residual flux may cause the gas to deteriorate.

The central wire, usually between 0.003 and 0.010 in. in diameter, must be free of points and pits. Use of a microscope for inspection is suggested. The wire must also be made of a material that will not pit easily in case of an electric discharge. Although tungsten is most commonly used, small diameter wires of stainless steel work very well (*39*) and are much smoother.

The inside surface of the tube, while not as critical as the central wire, should be machined smooth. Irregular features such as a side window can be introduced without affecting operation.

Filling gas. An inert gas, usually argon, plus a quenching gas is used to

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† Bibliography references 1 to 37 appeared with part I of this article, as did Figs. 1 to 5.

fill a G-M counter. Among the organic quenching gases, ethyl alcohol appears (*40*) more reliable than ethylene, although ethylene has the advantage that it is normally a gas at room temperature (*41*). Some commercially manufactured counters use nonorganic quenching gases that supposedly give infinite life. Generally, however, refilling becomes necessary prior to the expected life of even the organic quenching gas because of the thin windows used in spectrometer detectors. Thus organic quenching gases are thoroughly satisfactory.

Many commercial counters are filled to nearly atmospheric pressure. But since the counter gas in spectrometer applications is separated from the evacuated chamber by only a thin foil, it is desirable to have as small a pressure as possible within the counter. Conversely, if too low a pressure is used, the efficiency of the counter for electron detection becomes distinctly less than 100% since an electron may pass through the counter without producing an ion pair and thus escape detection. For energies below 150 kev it is possible to use a counter pressure as low as 2.0 cm Hg. For higher energies, the only tests known to the author indicate a lower limit of about 4.5 cm Hg to assure 100% efficiency. Also, the efficiency can possibly be a function of the type of filling mixture.

Thin windows. Mica or aluminum windows are seldom if ever made that transmit electrons of less than 35 kev. When studies of low-energy spectra are desired, foils made from the materials used for source backings (*13–16*) are usually employed. The final window is prepared by putting several layers of thin foil together to avoid pin-holes through which counter gas may leak.

Even with this precaution, the extremely thin foils required below 5 kev tend to leak slowly. To overcome any gradual loss of gas, the counter is kept at constant pressure by an automatic control such as a mercury valve (*36*) or a Cartesian manostat (*42*).

Loss of energy and scattering of electrons is the biggest problem with G-M counter windows. If the window is thicker than the range of the electrons, effectively no electrons pass through the window. If electron energy is much larger than the cut-off energy (the minimum electron energy for penetration of the window), essentially all the electrons will reach the counter's sensitive region. It is the intermediate region that is the nemesis of the beta-ray spectroscopist.

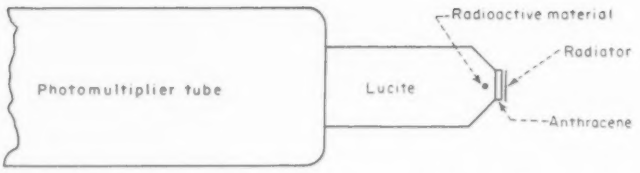

FIG. 7. Secondary-electron source, with anticoincidence technique, assures that only electrons originating in "radiator" are detected by spectrometer (66)

The rise in transmission between cut-off and the energy for 100% transmission is continuous, with the upper end being approximately an exponential function approaching 100% asymptotically as shown in Fig. 6. A close relation to electron scattering in matter exists (*11, 12, 43–45*), but most theories of electron scattering (*46–51*) have been developed for high-energy electrons. The problem of partial transmission in G-M counter windows arises in a much lower energy range.

Measurements of the relative positron and negatron transmission through aluminum and platinum have been made for comparison with scattering theory (*52*). In view of simplifying assumptions to put the theory into a suitable form for comparison, there is remarkable agreement between theory and experiment. It should be noted that the experiment required foil windows much thicker than those normally used in beta-ray spectroscopy.

The only known compilation of data for relative transmission coefficients for thin windows has been made by Saxon (*54*), who expressed his data as a correction factor that is the reciprocal of the transmission coefficient.

It should be re-emphasized that it is always desirable to measure 100% of the particles impinging on the surface of a counter window. The only valid reason for using partial transmission coefficients is if sufficiently thin windows cannot be produced. A valid test of transmission is to make measurements on two windows of different thicknesses. If the number of transmitted electrons is the same in both cases, it can be assumed that 100% of the impinging electrons are penetrating both windows.

Extremely thin foils stretched across the finite windows needed in spectrometer counters may not have the mechanical strength to withstand the pressure differential between the counter gas and the vacuum system. Grid supports can be used (*5, 33, 55, 56*), but the grid must be sufficiently thick and its edges properly shaped to prevent partial transmission of electrons through the grid.

An attempt has also been made to build a "windowless" spectrometer (*5*), but the need for a continuous supply of gas to the counter presented problems. The necessity of trapping all condensable vapors entering the spectrometer vacuum system limited the time during which a single set of data could be taken. Finally, the detector wasn't really windowless since a finite amount of gas remained between the sensitive volume of the counter and the vacuum chamber.

Accelerating potentials. Another attack on the window problem is use of accelerating potentials to give low-energy electrons enough added energy to penetrate foils of finite thickness. Such acceleration appears most feasible between the final spectrometer exit slit and the counter (*55*). But success has been limited by a large increase in background counting rate when voltages sufficient to produce 100% transmission of all real electrons are applied (*55, 57, 58*). The real counting rate is then only a small fraction of the background and does not lead to statistically accurate results.

A potential can also be applied at the source, but spectral distortions can be produced if electrons are not accelerated in their original direction of travel. One method uses a hemispherical source surrounded by a hemispherical grid to keep the electron emission symmetric in the forward direction (*59*).

At least two causes for the increase in background with accelerating potentials have been isolated: thermionic emission and ejection of secondary electrons. Townsend found qualitative agreement between the theory of thermionic emission and the number of low-energy electrons emitted when a copper sphere placed at the source position of a lens spectrometer was

heated (*60*). A cleaned tungsten filament, whose work function was much higher than the surface of the copper sphere, showed no appreciable emission until heated to a low red heat.

A large number of low-energy electrons are emitted when a radioactive source placed on a metallic sphere is used (*61*). The number is directly proportional to the source strength, decaying with it. The conclusion is that secondary electrons are ejected as beta particles pass through the source surface. Since such secondary electrons would probably have energies of only a few electron-volts (*62, 63*), apparent energy should equal applied voltage.

Additional spurious counts, occurring when voltage is applied to the source with the vacuum-measuring ionization gage in operation, can be attributed to ions escaping from the gage, being attracted to the source and producing secondary electrons (*60*).

No cause is immediately evident for the increased counting rates found when voltage is applied between the exit slit and counter. However, the solid angle subtended by the counter from the source is very small compared to the same solid angle from the exit slit. Thus a few cosmic rays producing secondary electrons in this region can cause an appreciable effect in addition to thermionic electrons escaping from spectrometer surfaces. Effects may also result from X-rays produced by accelerated electrons in one of the surfaces.

Scintillation Counters

A scintillation detector has been used successfully with lens spectrometers by several groups (*65–68*). Such a detector is made up of scintillator and a photomultiplier with its circuitry.

Since the pulse height produced by the photomultiplier is proportional to the energy of the electron incident upon the crystal, a pulse-height discriminator can be used in the detector circuit for energy selection in addition to the spectrometer energy selection. Zeldes, Brosi and Kettelle have used this technique to reduce the high-intensity low-energy background caused by secondary electrons produced by electromagnetic quanta penetrating lead shielding between source and detector (*65*). They were thus able to measure the low-intensity high-energy primary electrons.

Mann and Pearce have used two scintillation counters in anticoincidence to eliminate detection of electrons other than those produced in the radiator used in a study of low-intensity transitions in Ta^{182} and Sb^{124} (*66*). Their apparatus is shown in Fig. 7. Primary beta particles and Compton electrons generated in the Lucite and in most of the crystal leave an identifying pulse that is recorded by the source photomultiplier. Secondary electrons generated in the radiator leave no such pulse. An anticoincidence system between this source detector and the spectrometer detector prevents final recording of any pulse seen by both detectors, but those pulses seen only by the spectrometer detector will be recorded.

Jensen has used a lens spectrometer equipped with a scintillation counter and spectrometer to measure coincidences between gamma rays and their appropriate beta groups (*67*). This is a useful technique in studies of nuclear disintegration schemes. Siegbahn has built two lens spectrometers back to back for this same purpose (*69*). Fowler and Schuffler have built a coincidence spectrometer in which a single vacuum chamber and two detectors are used (*70*). This eliminates the problem of shielding one magnetic field from the other, but there is a problem in baffling to select electron energies.

Experimental problems. The sensitivity of photomultiplier tubes to magnetic fields requires that they be removed some distance from the spectrometer field and or shielded from stray fields. Mann and Pearce use a $\frac{1}{4}$-in. mild steel cylinder and end plate (*66*). Jensen uses a light pipe to extend the distance between the magnetic field and the photomultiplier (*67*). Wong and Lauritsen cover the photomultiplier with a magnetically compensating solenoid to eliminate the field in its vicinity (*68*).

Lucite or Plexiglas used for a light pipe should be cast rather then extruded. Strains and other irregularities in extruded material sometimes act to deter effective passage of light quanta. Also, the plastic should be very clear.

Effective passage of quanta along a plastic rod also depends upon good reflection at its walls. If angles of incidence cannot be made large enough to provide total reflection, a good reflecting material such as magnesium oxide should be packed around the exterior of the rod.

Dow-Corning 1,000-centistokes DC-200 silicone fluid encased in a Lucite tube has also been used as a light pipe (*71*), with results comparable to those obtained with a plastic rod. The chief gain is elimination of two reflecting surfaces.

At room temperatures photomultipliers produce a large number of low-energy pulses, called thermal noise, that are the major contribution to background. Lowering the tube temperature greatly decreases this noise (*67, 72*), allowing operation at lower energies. But the efficiency of the scintillator plus photomultiplier apparently does not remain near 100% for low energies (*67, 73, 74*), as shown in Fig. 6, and the scintillation counter thus does not circumvent all of the window difficulties of a G-M counter.

Electron Multipliers

Another detector for electrons is the electron multiplier (*75*), in which the electron beam enters directly into the multiplier, strikes the first electrode, and starts the avalanche ultimately observed by electronic equipment. Even though there is no window, the efficiency of an electron multiplier is far from linear for electrons (see Fig. 10 of *ref. 75*). It also depends greatly on prior treatment of the electrodes. It is often necessary to let air into a beta-ray spectrometer, and some method for treatment of electrodes under vacuum within the spectrometer is necessary.

While the electron multiplier has no distinct advantage over other, more simple detectors for electrons, it is good for detection of heavy particles and has been used for detection of recoil protons (*76, 77, 78*).

Magnetic Field

A critical factor in accurate determination of electron spectra is proper functioning of the spectrometer's magnetic field. Three requirements must be met for proper functioning:

1. The spectrometer must be designed and constructed so that the correct field exists in all regions of the electron path.

2. Field strength must be measured with an accuracy at least equal to the accuracy desired in electron momenta determinations.

3. The operator must be able to control the field so that during any single measurement it will remain constant to a degree equal to or better than the accuracy to which momenta are desired.

Experimental techniques bearing on these three requirements are discussed in the following sections.

Design and construction. When iron is used to locate the spectrometer's magnetic field, uniform density, especially near the pole face, must be maintained so that no measurable distortions will exist in the field shape. To keep the magnetic flux inside the iron yoke, the cross sectional area of the magnetic circuit should be large enough at all points, and the iron should be of low enough carbon content that saturation conditions are not approached.

The pole-face gap should be large enough to prevent the electron beam from approaching too close to the pole face. Within $^1/_{10}$ in. of the face, field fluctuations and discontinuities occur even with precise magnetics, but these fluctuations wash out rapidly with increasing distance from the iron (*79*). In a 40-cm radius-of-curvature magnet spectrometer (*55*), a region 1 in. high, centered in the median plane between pole faces that had many sharp edges, showed the same field strength at any horizontal level. Gap was 4 in.

In a lens or solenoid spectrometer that uses little or no ferrous material, the problem of establishing and shaping a magnetic field reverts to proper positioning of a set of current-carrying coils. The thin lens type uses a single series of coils half way between source and detector. The uniform solenoid type utilizes a single long coil in which source and detector are so placed that there is no gradient of field parallel to the walls of the cylindrically-shaped coils in the region between source and detector. Many more complex multiple-coil arrangements have been made to achieve a desired field (*1, 26*).

One important problem is elimination of that component of the earth's magnetic field perpendicular to the coil-produced field. This is usually done with large Helmholtz coils. An arrangement consisting of three coils also has been developed (*80*) for neutralizing the earth's magnetic field in and around a lens-type magnetic spectrometer.

Elimination of a vertical component of the field has also been effected by directing the axis of the spectrometer parallel to the earth's magnetic field (*81*).

Field measurement. In addition to being accurate enough for momenta determinations, measurement must be accomplished in a short enough time to represent a true measure of the magnetic field at the time a single portion of the electron spectrum is studied.

One of the most common tools for field measurement is the flip coil and ballistic galvanometer. A coil with a large number of turns of fine wire is mounted in the magnetic field with plane of the coil perpendicular to the field direction. This coil is rotated quickly through 180 degrees. The voltage thus induced across the coil causes a certain quantity of charge to flow through the circuit consisting of the flip coil and the galvanometer.

Three things can cause difficulties in this type of field measurement. One is the time required for the galvanometer to return to zero before another reading can be made. If this is made too short, the second difficulty arises, namely, making sure that the time to flip the coil is short compared to the galvanometer period. Too short a galvanometer period produces a spread in readings dependent upon how fast the flip coil is operated. Some compromise between these two items is usually arranged by adjusting series and shunt resistors to critically damp the galvanometer.

A third error lies in reading the galvanometer; 0.2 mm is the limit of accuracy on the usual scale, and a large scale deflection is required. But the time the galvanometer remains at its maximum deflection diminishes as the deflection is increased. A compromise is again needed.

The time required for magnetic-field measurements has been greatly reduced by use of a continuously rotating coil. Such a device has been constructed (*82*); its speed is kept constant by a synchronous motor. The coil's output current is then a measure of the magnetic field strength.

A modification of this method eliminates any difficulty caused by varying contact resistance of brushes. A pair of transformers are used, both of which have fixed secondaries connected to a measuring device (*83*). The primary of one transformer rotates on the same shaft as the continuously rotating coil within the spectrometer field. The alternating current produced in the secondary of this transformer is mixed with the a-c signal from the secondary of the second transformer, whose primary coil rotates in a known field produced by a pair of Helmholtz coils. By adjusting the field produced by the Helmholtz coils, the exact spectrometer field is determined by a null method. Measurements to better than 1 part in 1,000 can be made.

The difficulty with this system is that it requires two identical synchronous motors operated 180 deg out of phase to drive the coils. Thus any change in external frequency changes the speed of both motors at the same rate and produces no change in output signal. In a similar system, shown in Fig. 8, both coils are placed on one shaft and oriented to provide voltages 180 degrees out of phase (*84*). This eliminates the need for synchronous motors, and the system can make magnetic field measurements to five significant figures.

While primarily applicable to measurement of uniform magnetic fields, a method employing nuclear resonance absorption appears to be the most accurate. The radio frequencies used can be measured to many significant figures. Measurements of momentum, $B\rho$, can be made using these techniques after the spectrometer is calibrated in terms of applied frequency, ν. Calibration is possible if two different internal conversion lines from the same nuclear transition can be measured in the spectrometer (*86*). Lindström uses the K and L_I conversion electrons from ThB. Since X-ray transitions are generally known to at least five significant figures, the energy of separation, ΔW, between two such lines can generally be expressed to the same accuracy. In terms of the electron momenta p_1 $(=AH_1\rho)$ and $p_2(=AH_2\rho)$, respectively, for the two conversion electrons, A being a constant of proportionality

$$W = (p_1^2 + 1)^{1/2} - (p_2^2 + 1)^{1/2}$$
$$= [(AH_1\rho)^2 + 1]^{1/2} - [(AH_2\rho)^2 + 1]^{1/2}.$$

Dividing by $H_2\rho$

$$W/H_2\rho = [(H_1/H_2)^2A^2 + 1/(H_2\rho)^2]^{1/2} - [A^2 + 1/(H_2\rho)^2]^{1/2}$$

All quantities are known or can be determined experimentally with an accuracy of at least five significant figures, with the single exception of the quantity $H_2\rho$. Since any other field H_n will produce a resonance at a frequency ν_n, any electron momentum can be measured by the ratio $H_n\rho/H_2\rho = H_n/H_2 = \nu_n/\nu_2$.

This method of measuring magnetic fields has also been used by DuMond *et al.* (*33*).

Field control. Adequate control of the spectrometer's magnetic field is

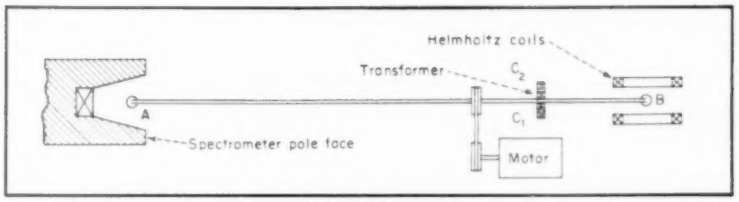

FIG. 8. Apparatus used for measuring magnetic field in spectrometer. Current within Helmholtz coils is adjusted to produce null with transformer secondary C_2 that does not rotate. Detector coils A and B and primary C_1 rotate on common shaft

especially important if high resolution is desired. Accuracy to one part in several thousand may be needed during taking of a single set of data.

An iron core spectrometer usually needs only a small current to produce the magnetic field; thus a modification of a constant-current circuit designed by Elmore and Sands (*87*) can be used successfully as an electronic control. Hysteresis in the iron core of the spectrometer, however, prevents returning to a previous magnetic field by merely adjusting the control to its previous current.

Instead of the resistance steps of Elmore and Sands original circuit (Fig. 7.14 of ref *87*), three high-voltage rheostats (such as Ohmite model K) are used to provide the necessary continuity for setting the spectrometer to any magnetic field. Grid bias for the 807 and 6AG7 tubes has been provided by small B batteries. Larger batteries supply the 1B4P plate circuit to provide long life. Filament current for the 1B4P should be supplied by a rechargable wet cell. We have found Willard types ER 24-2 or ER 34-2 plastic nonspill cells convenient.

Grid bias on the 1B4P must be constant also; extreme accuracy is needed for the system to work properly. Mallory mercury cells have a flat voltage characteristic at 1.345 volts over almost their entire life (*88*).

The most common source of instability in this circuit is loose elements in the 6AG7 or 807 tubes. A good check for loose elements, which may not be detected with a normal tube tester, is to tap the tube with a pencil while it is in place in the operating circuit. The magnet current should be set at some readable current; fluctuations will occur if a grid element is loose in the tube tapped.

Lens and other spectrometers not utilizing an iron yoke require larger currents to produce the magnetic field. With other conditions constant in the spectrometer region, the field is directly proportional to the current through the spectrometer coils plus a small additive constant attributable to the earth's magnetic field.

The current for these spectrometers has generally been provided by motor-generator sets whose field-coil currents are regulated by some sort of feedback from the magnet current. Although designed for nuclear magnetic resonance measurements, a typical current stabilizer is that described by Sommers *et al.* (*89*). This device measures the direct current from a standard resistance and feeds it back through an a-c converter to an amplifier whose output controls the field current in the motor-generator set.

Proton resonance can also be used to control the output from the motor-generator set. DuMond has described such a system (*90*), with a feed-back system essentially the same as that described by Sommers *et al.* (*89*). However, the error signal comes from the signal produced by the proton resonance inside the magnetic field. The control thus cannot have an error because of nonlinear relationships between coil current and magnetic field. This type of control can be used on all types of spectrometers, regardless of the amount of iron in their structures. About the only limitation is that the resonance head must be located in a region of uniform field strength to provide a sharp proton resonance.

* * *

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The Biological Effectiveness of Thermal Neutrons on Mice—I

To determine the biological effectiveness of thermal neutrons, mice were exposed in the Los Alamos "Water Boiler" thermal column. Before biological measurements could be made, the dose contributed by the column's inherent gamma contamination was determined by means of methods described in this first part of the article

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THE PRIMARY PURPOSE of the present experiment was to determine the relative biological effectiveness of thermal neutrons by comparing mice exposed in the thermal column of the Los Alamos homogeneous reactor with mice exposed to X-rays. Since the thermal column of this reactor (*1*) had not been previously used for biological exposures, it was necessary to investigate the physical properties of the column.

Several theoretical calculations concerning the biological effectiveness of thermal neutrons have appeared (*2–4*), but all such theoretical treatments necessarily involve certain assumptions in which there are large uncertainties.

To date, only two attempts (*5, 6*) have been made to measure the effectiveness of thermal neutrons on mammals. On both occasions, severe dose-measurement problems were encountered.

In this first part of the article, the characteristics of the thermal column's neutron flux and inherent gamma flux are discussed.

North Thermal Column

Figure 1 is a schematic of the north thermal column viewed toward the west. It shows the 8½-in. bismuth wall, which attenuates γ-rays produced in or near the sphere; the graphite column or chimney, which moderates the fission neutrons to thermal energy; and the cavity in the graphite in which the biological exposures and physical measurements of this experiment were made. During exposures, physical detectors and animal groups were centered at point *P*.

Figure 2 shows the end of the thermal column at the north face of the pile. The 7-ton plug was pulled out on rails

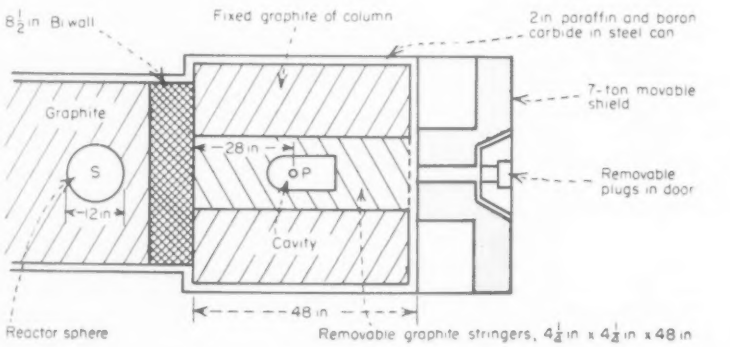

FIG. 1. Essential features of north thermal column of homogeneous reactor

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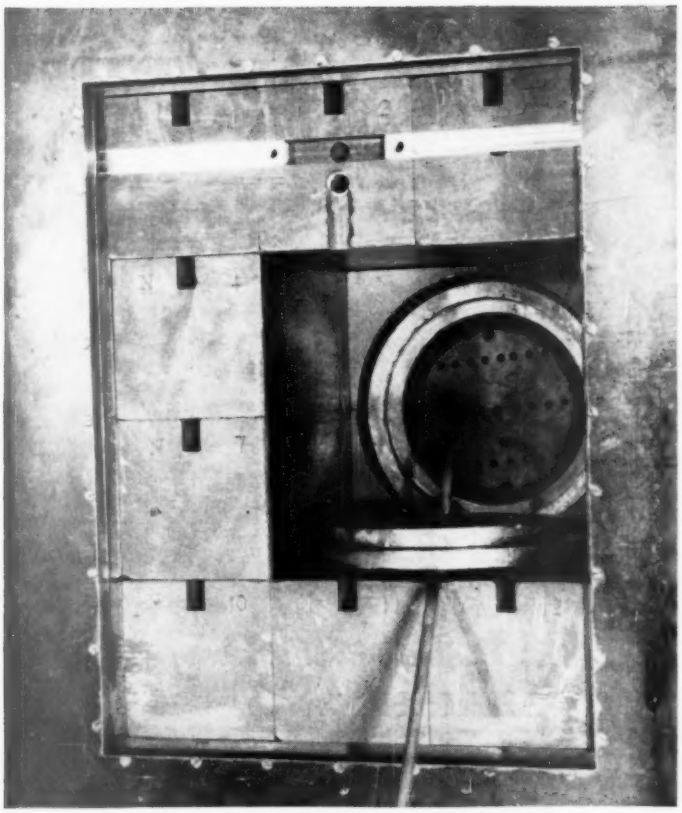

FIG. 2. Thermal column showing exposure cavity with Bi shield and graphite exposure cage

to insert mice or radiation detectors for exposure. After running the reactor at 25 kw, it was necessary to allow about 5 min for "cooling" of the reactor before the plug was rolled out.

The thermal column was constructed to provide a maximum neutron-to-γ-ray ratio. The neutron flux anywhere in the column is directly proportional to the rate at which fissions are taking place in the sphere.* Residual gamma activity, due to the build-up of beta-gamma emitters during reactor operation, made no significant contribution to the dose received by the mice. [Those not familiar with the design of thermal columns are referred to a recent text (7).]

Thermal-neutron flux measurements were made with both In and Mn foils. The In foils were calibrated in a sigma pile at Los Alamos. The Mn foils had been calibrated independently in terms of the number of fissions produced in U^{235}, the number of counts registered by a well-defined volume of a BF_3 counter, and the known thermal-neutron cross sections of boron and U^{235}. The measured values for the neutron flux at 25 kw using these foils were $2 \pm 0.20 \times 10^{10}$ n/cm^2/sec for In and $1.7 \pm 0.17 \times 10^{10}$ n/cm^2/sec for Mn. In this article, the weighted average value of $1.8 \pm 0.18 \times 10^{10}$ n/cm^2/sec is used.

Correction factors. Three major correction factors are required before a correct value can be derived for the biological effectiveness of the neutrons:

1. γ_c, inherent gamma contamination in the column—the mortality of mice exposed in the column was due in part to the inherent gamma flux. As anticipated from previous experiments (*5, 6, 8*), most of our effort was expended on this problem.

2. γ_i, intermouse gamma dose—a mouse in a neutron flux becomes a gamma source. This factor was not difficult to evaluate once the problem of inherent gamma dose was solved.

3. ξ, attenuation of the thermal neutron flux in a compact group of mice—a mass of animals acts as a neutron "sink" and depresses the flux within the group of mice.

The chief purpose of the present experiment was to determine F, the number of thermal neutrons/cm^2 that, when incident upon a single mouse, will, in terms of 30-day lethality, produce a biological effect equal to that evoked by 1 r of 250-kvp X-rays. This quantity is

$$F = (nv)\xi/(R - \gamma_i - \gamma_c)$$

where nv = the thermal neutron flux (n/cm^2/sec) in the empty graphite cavity, R = the lethality response in rem/sec observed in the mice, γ_c = the dose in rem/sec due to the inherent gamma contaminant in the column, γ_i = rem/sec of intermouse gamma dose, and ξ = a fraction less than one, which takes into account the attenuation of the thermal flux in a compact group of 15 mice.

Measurement of γ_c

Unshielded gamma detectors, e.g., film and ion chambers, when placed in a thermal column, become activated, and their response is uninterpretable in terms of radiation dose.

Boron, which has little direct absorption effect on the γ-rays of the thermal column and is opaque to thermal neutrons, has been used as a shield several times (*5, 8, 9*). However, in the reaction $B^{10}(n,\alpha)Li^7$, 94% of the captures result in an excited state of Li^7 and the emission of a 0.48-Mev photon.

In 1944, Zirkle and Raper suggested that, on theoretical grounds, metallic Li would be the best ion-chamber shield. This suggestion was followed in the present experiment.

Thermal neutron reactions in lithium. The efficiency of the lithium shield in absorbing slow neutrons results from the following reactions and characteristics:

$Li^6(n,\alpha)H^3$, σ_c = 860 barns, no accompanying gammas

$Li^7(n,\gamma)Li^8$, σ_c = 0.033 barns, 1.98 Mev total γ-ray energy

Neutron capture in Li^7 produces a high-energy gamma photon, and Li^7 is more abundant than Li^6 (92.5% vs. 7.5%) in natural lithium. The large difference in cross sections, however, results in only one capture in 2,000 taking place in Li^7. Consequently,

* The radiation characteristics of the column have been studied in detail by the members of Group P-2 (LASL).

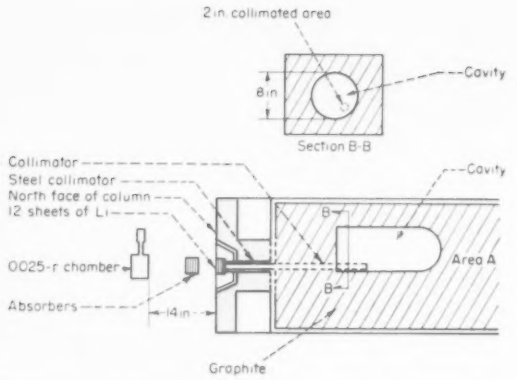

FIG. 3. Arrangement for studying absorption characteristics of inherent gamma flux in good geometry (Al absorbers at center, Li absorbers in port)

relatively few capture γ-rays are produced.

The reading on the r-meter inside the lithium, therefore, is due to γ-ray contamination inherent in the thermal column.

Good Geometry Studies

Before attempting to interpret radiation measurements made inside the thermal column, studies were made of the mixed radiation of the column under conditions of good geometry.

A beam was led out of the north face of the pile (Fig. 3). The r-meter wall thickness was increased to $\frac{7}{8}$ in. to maximize the chamber readings. The data of Fig. 4 were obtained using the ion chamber and lithium absorbers shown in Fig. 3. Following exposure with no lithium absorber interposed in the beam, $\frac{1}{2}$-hr observation showed no drift due to activation of the chamber materials.

The curve in Fig. 4 may be resolved into two straight-line components by first drawing a line through the last six points. Subtracting this straight line from the curve gives a second curve, the slope of which over one decade indicates an absorption coefficient $\mu_2 = 2.42$ cm^{-1}. The most obvious interpretation is that the hard component is a gamma beam and the soft component is a slow-neutron flux.

With the accepted cross section for thermal neutrons in lithium, it can be calculated that the effective average energy of the neutron component is 0.052 ev ($\sigma_{total} = 52$ barns or $\mu_2 = 2.42$ cm^{-1}).

The slope of the hard-component curve is very flat, indicating a high average energy.

Identification of soft component. With the arrangement of Fig. 4, a B^{10} counter* was substituted for the 0.025-r chamber. The counter circuit was biased to avoid counting γ-rays.

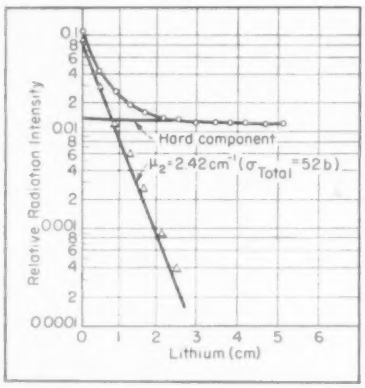

FIG. 4. Absorption of thermal-column mixed radiation in Li in good geometry

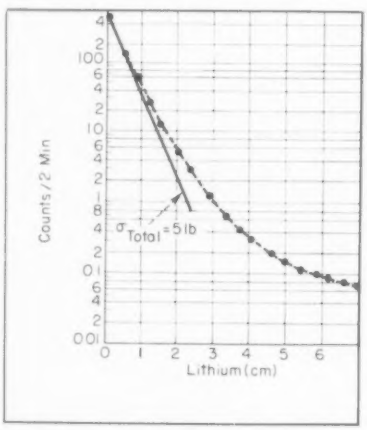

FIG. 5. Absorption of neutrons in good geometry using biased B^{10} counter

The lithium absorption curve was repeated with this counter, and the data are given in Fig. 5. The wide variation of the power level of the pile was necessary because the maximum counting rate that could be tolerated by the counter was about 250 cpm. Counter background was of the order of 0.5 cpm.

The slope of the absorption curve (Fig. 5), determined by only the points in the first decade, indicates a slow-neutron energy of 0.052 ev ($\sigma_{total} = 51$ barns). Investigations of the neutron flux in the column performed by Group P-2 of LASL showed that the neutrons at this point in the column are thermalized to 22° C. It is highly probable, therefore, that the soft component ($\mu_2 = 2.42$ cm^{-1}) indicated by the absorption curve shown in Fig. 4 is a thermal-neutron flux.

Identification of hard component. Further good-geometry absorption studies were made with the 0.025-r chamber and Li, C, Al, Fe, Bi, and Pb absorbers.

Data for all the absorbers were obtained with the 12 original Li absorbers (5.04 cm) in place. To obtain the Li curve, Li was added to a total thickness of 20 cm. The curves are shown in Fig. 6. In fitting the curves, only the first three points of each were considered.

The following effective average γ-ray energies (Mev) are indicated: Li, 2.3; C, 3.5; Al, 3.7; Fe, 4.0; Bi, 2.4 or 4.9 (minimum in μ_{total} near 3 Mev); Pb, 2.4 or 4.9 (minimum in μ_{total} near 3

* The writers are indebted to J. W. Starner and W. E. Stein of Group P-2, LASL, for supplying this equipment and for assistance with this part of the experiment.

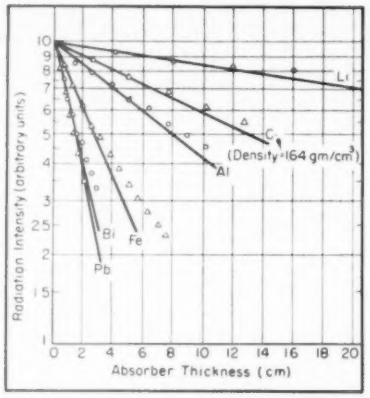

FIG. 6. Absorption of inherent γ-rays in various absorbers in good geometry

Mev). These curves establish the hard component as a gamma beam having an effective average energy of about 4 Mev.

Effect of ion-chamber wall thickness. As a further check on the quality of the inherent gamma radiation of the thermal column, the arrangement shown in Fig. 3 was utilized, leaving 12 Li absorber plates in place in the port. Instead of the absorbers previously used, the wall thickness of the ion chamber was varied with the acetate caps shown in Fig. 3 surrounding the 0.25-r chamber.

With this arrangement, the maximum reading was obtained with a wall thickness of 1.86 gm/cm² (added acetate = 1.7 gm/cm²). As shown by Evans (*10*) and Gray (*11*), a maximum reading is obtained when the chamber-wall thickness is about equal to the range in the wall of the maximum-energy electron produced by the gamma beam. The results in the present case indicate a gamma beam having an effective average energy of approximately 4 Mev.*

As supporting evidence, it is important to note that the dominant thermal neutron capture reaction in graphite produces a 4.9-Mev γ-ray (*12*).

In Fig. 4, the extrapolation of the hard component to zero Li absorber shows that the gamma flux is responsible for $(0.0143 \times 100)/0.1056 = 13.5\%$ of the total discharge observed.

TABLE 1—Absorption of Thermal-Column Neutrons in Li in Good Geometry

Layers of Li	cm	Counts/2 min/60 watts*			
		60 watts	600 watts	6,000 watts	25,000 watts
0	0	514			
1	0.43	157			
2	0.82	60.5			
3	1.22	26.5	26.4		
4	1.59	12.9	12.5		
5	2.08	5.0	5.2	5.03	
6	2.46		2.8	2.72	
7	2.92		1.28	1.30	
8	3.39		0.66	0.65	
9	3.75			0.45	
10	4.12			0.30	
11	4.61			0.20	
12	5.04			0.15	0.15
13	5.41			0.10	0.12
14	5.80				0.10
15	6.21				0.090
16	6.60				0.079
17	7.01				0.068

* All results are normalized to a power level of 60 watts.

Poor Geometry Studies

Twenty cylindrical and bullet-shaped lithium shields (Fig. 7) were designed to make poor-geometry (pseudo-spherical) measurements of the inherent gamma dose in the exposure cavity within the column. Victoreen 100-r chambers were used; the dose-rates were such that half-scale readings could be obtained in 1–2 min exposures.

Source of inherent gamma flux. Since thermal neutrons at any point in the thermal column are effectively random in direction, all radiations to which the neutrons give rise are random in direction. Therefore, the radiations entering the cavity from all areas of the wall are qualitatively similar. If, how-

* Recent studies by Bob E. Watt, Group P-2 (LASL), have shown that the gamma flux of the south column has a maximum energy of 4.9 ± 0.1 Mev.

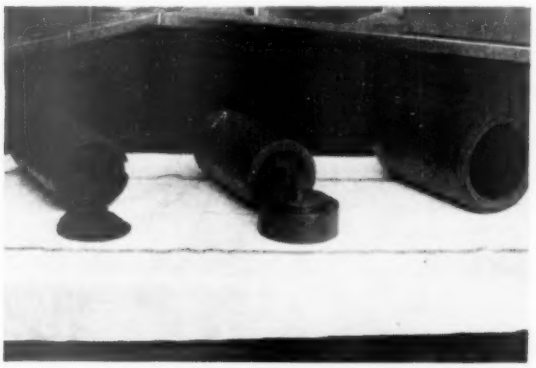

FIG. 7. Back-end view of cylindrical, bullet-shaped, and outer Li shields

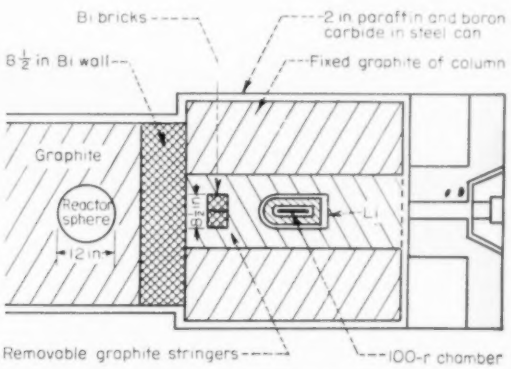

FIG. 8. Arrangement used to study source and direction of inherent gamma flux

FIG. 9. Graphite exposure cavity showing Bi shells and graphite mouse exposure cage

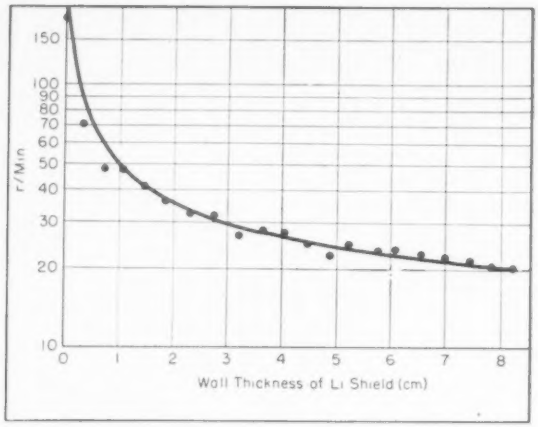

FIG. 10. Absorption of mixed radiation in Li in poor geometry

ever, any appreciable fraction of the inherent gamma flux of the thermal column is due to fission γ-rays originating in the reactor sphere (Fig. 1), then the good-geometry setup of Fig. 3 would give a biased picture of the cavity γ-rays since the steel collimator pointed directly toward the sphere.

With a bullet-shaped Li shield (Fig. 7) of 3.07-cm wall thickness, the arrangement shown in Fig. 8 was used to study the source and direction of the inherent γ-rays. Two high-purity bismuth bricks, each $4\frac{1}{4} \times 4\frac{1}{4} \times 8$ in., were placed crosswise in the column at a point between the reactor sphere and the r-meter encased in the Li shield.

The reactor was operated at 25 kw and an r-meter reading taken with the Bi bricks in place. The bricks were replaced with graphite, and another reading was taken. The respective readings were 31.5 and 31.7 r/min.

These results eliminate the possibility that a significant fraction of the inherent gamma flux is due to γ-rays from the reactor sphere penetrating the permanent $8\frac{1}{2}$-in. Bi wall. In addition, they support the conclusion that the inherent gamma photons enter the cavity from all directions. Since the two Bi bricks shown in Fig. 8 subtend a small fraction (0.04) of the full 4π solid angle, they should make little difference in the reading if the γ-rays are random in direction. Also the graphite intervening between the Bi bricks and the ion chamber would further mask the effect of the Bi bricks as far as graphite capture γ-rays are concerned.

Mixed radiation in graphite cavity. It is now possible to construct a qualitative picture of the mixed-radiation flux in the exposure cavity.

The thermal neutrons are essentially random in direction. The dimensions of the air-filled cavity are such that the number of neutrons undergoing reactions with air is negligible. Hence, in the empty cavity all neutrons that emerge from the wall travel in straight lines until they reenter the wall. Since the thermal neutron flux falls off in moving down the column away from the source, more neutrons enter the cavity from the hemispherical end than from the flat end. The difference is about a factor of 3 per foot; therefore, the axis of the graphite mouse cage (Fig. 9) should not exceed 4 in.

The capture gamma source strength in any region of the graphite is proportional to the neutron flux in that region, and consequently there are more gamma photons entering the cavity from the hemispherical end than from the flat end. The photons are random in direction.

The picture outlined implies that all of the photons traversing the graphite cavity originate as graphite capture γ-rays. Although this is highly probable, no positive experimental evidence has been adduced to prove it.

Li extrapolation method. The center of the ion chamber was located at P in Fig. 1. Twenty different thicknesses of Li shield were available.

At the outset, the bullet-shaped shield (Fig. 7) was placed inside the six outside shells. This assembly had an outer diameter of 8 in. and just fit the graphite cavity. Seven 2-min readings were obtained with the 100-r chamber at a power level of 25 kw by successively removing the shells until only the solid shield remained (see Fig. 10).

To obtain neutron absorbers of lesser thickness, the solid lithium shield was replaced by a laminated cylindrical shield (Fig. 7) consisting of 13 concentric, coaxial, hollow cylinders made of sheet lithium. Each cylinder consisted of a sleeve and two loose-fitting ends. Assembly was accomplished by molding the malleable lithium by hand. During exposure, the shield and r-meter were centered in the cavity by means of graphite wedges.

Readings were taken as successive layers of shield were peeled off (Fig. 10). In addition to the r-meter readings, seven neutron-flux measurements with Mn foils were made as the layers of Li metal were removed (see Fig. 11).

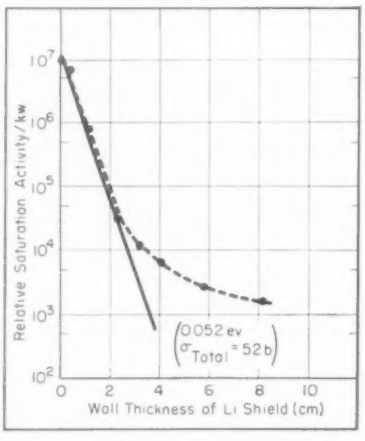

FIG. 11. Activation of Mn foils inside Li shields in poor geometry

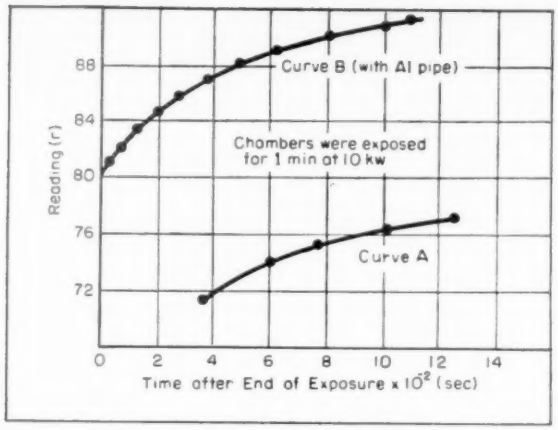

FIG. 12. Drift in 100-r chamber due to neutron-induced chamber activation

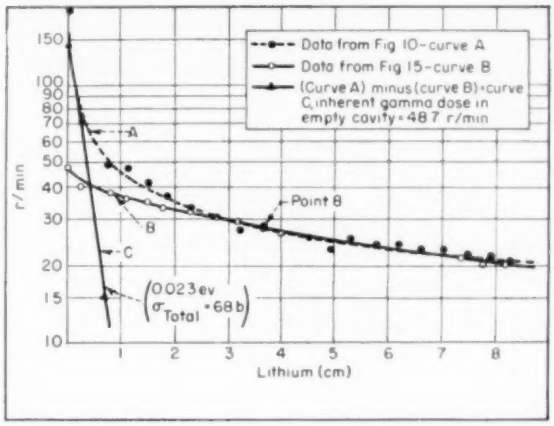
Data from Fig 10-curve A
Data from Fig 15-curve B
(Curve A) minus (curve B)=curve C, inherent gamma dose in empty cavity=48.7 r/min
Point B
r/min
Lithium (cm)

FIG. 13. Resolution of poor-geometry Li absorption curve into hard and soft components

In both Figs. 10 and 4, the initial portion of the curve has the steep slope of thermal-neutron absorption followed by a portion with relatively little slope.

Extrapolation of Fig. 10 to zero lithium absorber indicates that about 16% (32 r/min) of the total reading, following exposure of a 100-r chamber with no shield, is due to inherent γ-rays. This indicates that the ratio of gamma dose rate to thermal-neutron flux is about the same in poor and in good geometry.

However, the curve of Fig. 10 cannot be resolved into two simple elements as can the curve of Fig. 4. In poor geometry, the quantity of the radiation reaching the ion chamber is not necessarily a simple function of any dimension of the shield employed.

Activation of chamber materials. When exposed in poor geometry with no lithium shield, the chamber acquired a visible drift due to induced activity. This made a correction necessary.

Chambers were exposed in the thermal column for 1 min with no Li shield at a power level of 10 kw. After their removal, the drift was followed for several minutes. The data are presented in Fig. 12. The first point of curve *A* was obtained after a 360-sec cooling period.

To evalute the importance of drift, it was necessary to extrapolate backward in time to the end of exposure. To minimize the extrapolation, a special study was made with the r-meter held in the cavity in the end of a long aluminum pipe. After a 1-min exposure at 10 kw, the chamber was quickly withdrawn from the reactor by means of the pipe. The first reading was obtained 27 sec after the end of exposure. The chamber remained in the electrometer and the drift of the fiber recorded (curve *B*, Fig. 12).

The absolute readings of curve *B* are higher than those of *A* because of neutron capture in the end of the Al pipe, which was in the cavity. However, the drift was about the same for both chambers and amounted to 8 r in 6 min (corresponding to 20 r in 6 min when running at 25 kw). The data of Fig. 10 are replotted as curve *A* in Fig. 13 after correction for chamber activation.

With one or more layers of lithium shield around the chamber, the total error due to acquired drift was negligible.

Hard component in poor geometry. Three centimeters of Li is sufficient to make the r-meter reading due to thermal neutrons negligible. Presumably, therefore, the flat portion of curve *A* in Fig. 13 is due to inherent gamma flux.

In comparison with Fig. 4, the slope of this portion of curve *A* is too great to be accounted for by the absorption of inherent γ-rays in the Li shield. Moreover, in poor geometry the absorption of inherent γ-rays in lithium should be even less apparent than it is in good geometry because Compton scattering is partially compensated (i.e., for every photon scattered away from the chamber, another of lesser energy is scattered into it).

The effective absorption coefficient for 4-Mev gamma photons in this pseudo-spherical geometry should be about equal to σ_a, the absorption por-

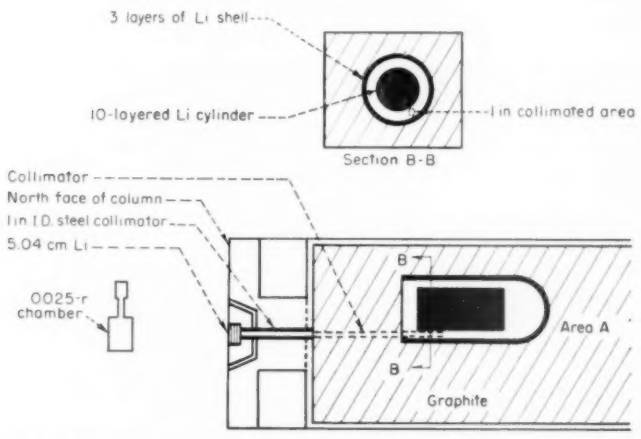

FIG. 14. Arrangement to study effect of Li in cavity on inherent gamma flux in cavity

tion of the Compton coefficient. This approximation is justified since the detector is measuring roentgens (*10*). For 4-Mev gamma photons in Li, $\sigma_a = 0.0094$ cm^{-1}. It may be seen in Fig. 10 that the scatter of the points is such that a slope of 0.0094 cm^{-1} would not be discernible (total attenuation in 8 cm = 7%). The last thirteen points of this curve actually indicate an absorption coefficient of about 0.055 cm^{-1}. So the absorption of inherent γ-rays in lithium cannot account for the slope of the tail portion of the curve in Fig. 10.

Neutron "sink" effect of shield. The only other possibility for explaining the slope of Fig. 10 is the neutron sink effect of the Li shield. This effect reduces the neutron flux around the cavity and therefore reduces the number of capture γ-rays originating in that region of the graphite.

The neutrons diffusing through the thermal column may be divided into two categories: (1) neutrons that enter the cavity one or more times before being captured (cavity neutrons), and (2) neutrons that never enter the cavity (noncavity neutrons).

Similarly, the gamma photons in the thermal column may be designated as cavity and noncavity photons. The flux of cavity photons is synonymous with the inherent gamma flux. The latter may be subdivided as: (1) photons created by the capture of cavity neutrons (cavity-neutron gammas) and (2) photons created by the capture of noncavity neutrons (noncavity-neutron gammas).

The usefulness of these terms lies in the fact that cavity neutrons, and hence cavity-neutron γ-rays, are influenced by neutron capture in the lithium shield (sink effect). On the other hand, the flux of noncavity-neutron γ-rays entering the cavity cannot be influenced by the lithium.

The experimental arrangement used to demonstrate the reality of the lithium neutron sink effect involved a return to good-geometry conditions (Fig. 14). A collimated beam of cavity γ-rays was taken from a well-defined area on the hemispherical end of the graphite cavity. The beam passed along side of, but did not impinge upon, the 10 layers of cylindrical Li shield located in the center of the cavity. The three outside layers of Li shield were *not* present. The beam was led out through the face of the reactor and filtered through 5.04 cm of Li, and the γ-radiation measured with a 0.025-r chamber. As individual layers of the Li shield were removed, the gamma flux measured by the r-chamber steadily increased (the cavity-neutron gamma flux increases while the noncavity-neutron gamma flux is not altered). These data, comprising the first 11 points of Fig. 15, show that the sink effect of the shield does indirectly decrease the flux of cavity-neutron γ-rays. The slope of the curve in Fig. 15 is due only to the sink effect.

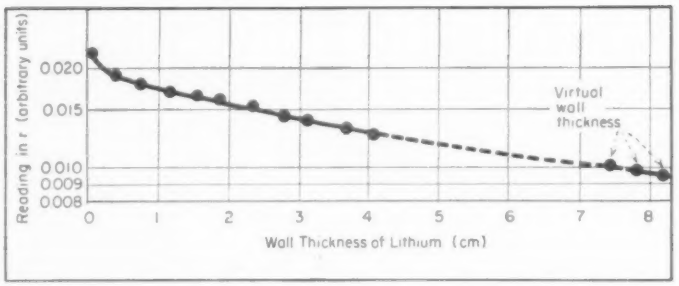

FIG. 15. Indirect effect of Li in cavity on inherent gamma flux in cavity

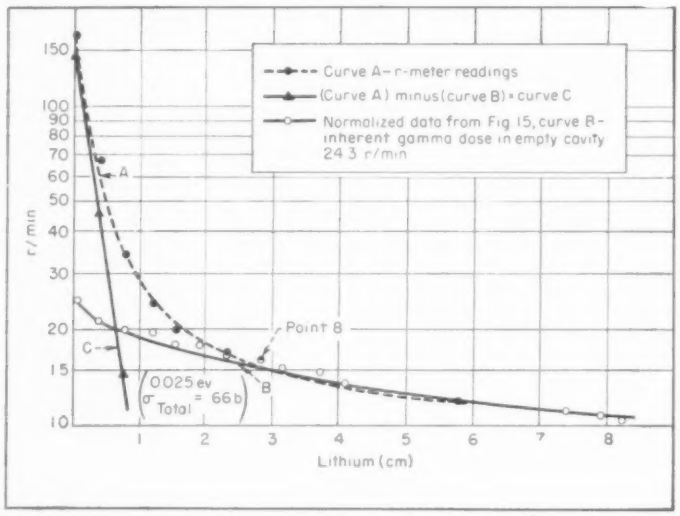

FIG. 16. Resolution of poor-geometry Li absorption curve, with Bi shell present, into hard and soft components

Considering Curve *A* of Fig. 13, the shape of the curve can be the result of no more than three mechanisms: (a) neutron capture in the lithium; (b) absorption of inherent gamma in the lithium shield; and (c) the indirect decrease in gamma flux seen in Fig. 15, which is due to the neutron sink effect of the lithium.

If, as we have said, (b) is unimportant, then two things should follow: (a) the slope of the tail portion of curve *A*, Fig. 13, should match the slope of the curve in Fig. 15; (b) when the curve in Fig. 15 is normalized and subtracted from curve *A*, Fig. 13, the new remainder curve obtained should be the absorption curve for thermal neutrons in lithium.

The subtraction suggested in (b) has been made in Fig. 13. The slopes of curves *A* and *B* match reasonably well, and the slope of curve *C* indicates an effective average energy of 0.023 ev as compared with 0.025 ev for thermal neutrons.

In making the point-by-point extrapolation subtraction in Fig. 13, curves *A* and *B* were normalized at point 8 (wall-thickness of lithium 2.77 cm). This wall thickness was chosen because of the following considerations:

The first 11 points of Fig. 15 were obtained with the same 10 cylindrical Li shields that were employed to get the first 11 points of curve *A* in Fig. 13, the difference being, of course, that the data of Fig. 15 were taken with a

0.025-r chamber outside the pile in good geometry, while the data of Fig. 13 were taken using a 100-r chamber inside the Li shield in poor geometry. Two differences should be noted between these two physical situations.

1. In poor geometry, the 100-r chamber begins to show the direct effects of thermal neutrons when the Li shield thickness is reduced to less than 2.77 cm Li (Fig. 4). On the other hand, at no time do thermal neutrons reach the 0.025-r chamber in the experiment in good geometry (Fig. 15).

2. To reach the 100-r chamber in the poor geometry experiment, all cavity γ-rays must pass through *varying* thicknesses of Li in the cylindrical shield. In the good geometry case, Fig. 14, the γ-rays traverse a constant thickness of lithium (5.04 cm).

By normalizing at point 8, the importance of the two physical differences just noted is minimized because, with wall-thickness of 2.77 cm of Li, the number of neutrons reaching the 100-r chamber in poor geometry is negligible. Also, with this thickness of lithium, the absorption of inherent gamma rays in the cylindrical shield in poor geometry is as small as possible.

The intercept of curve *B*, Fig. 13, with the ordinate gives a value of 48.7 r/min for the inherent gamma dose rate in the empty graphite. Obtaining this number is the whole point of the lithium extrapolation method, and comparison should be made with the rough estimate of 32 r/min made previously.

As may be seen in Fig. 14, section *B-B*, the use of a collimated beam in an off-center position allowed as many as 10 layers of cylindrical Li shield to be placed in the cavity. It was also possible to place as many as three layers of Li shield *outside* of the 1-in. collimated beam.

Since the opportunity presented itself, the 10 cylindrical layers of Li were removed, and the three outer shells put in place. Readings were then taken on the 0.025 r-chamber, as the three large Li shells were removed, beginning with the outermost first. The first result, 0.0094 (Fig. 15), was obtained with only the three outer shells in place. With this arrangement, all of the cavity neutrons are captured and the cavity-neutron gamma flux must be zero. Therefore, the reading of 0.0094 must be due entirely to noncavity-neutron gammas. With no lithium in the cavity, the reading was 0.0221. It appears, therefore, that the inherent gamma dose in the cavity consists of 57% cavity-neutron gamma and 43% noncavity-neutron gamma.

This division applies, of course, only to the particular graphite cavity employed. It is conceivable that spherical graphite cavities of varying radii might yield a novel type of information concerning the neutron and gamma fluxes in the column.

Since even thin sheets of lithium are relatively opaque to the neutrons in the column, a thin-walled shell should be just as effective a gamma depressor as the same shell with its center filled with lithium. For the last three points of Fig. 15, the value for the abscissa in each case is not the actual wall thickness of the shell used, but rather the thickness that would be obtained if the given shell were filled with the maximum amount of lithium possible (called virtual wall-thickness). These last three points in Fig. 15 appear to connect smoothly with the first eleven, indicating that the role of the lithium has been correctly interpreted.

Returning to the first eleven points of Fig. 15, the reason for the change in slope between the first and second points may be seen in Fig. 7, which shows the laminated lithium shield. From the photograph, it seems plausible that the removal of successive individual shells should cause similar decrements in the neutron sink effect. But the smallest cylindrical shield must still be large enough to accommodate an r-chamber. Consequently, in going from the smallest lithium shield to zero lithium, a break in the curve of Fig. 15 is to be expected.

It should also be pointed out here that in Figs. 13 and 15 the abscissae are, in both cases, arbitrary. That is, there is no *a priori* reason for believing that the sink effect of a given lithium shield will be a simple function of the wall-thickness of the shell. Nor should the sink effect, for that matter, be a simple function of any dimension of the shield. Actually, the sink effect probably bears some complex relationship to the total surface area of the shield, but the important point is that the exact nature of this relationship need not be known to make the point-by-point extrapolation subtraction used in Fig. 13. Whatever the physical realities may be, plotting the abscissae in terms of wall thickness represents a coordinate transformation that must be the same in Fig. 13 as it is in Fig. 15.

Since the completion of the experiment it has been pointed out that in Fig. 14 the cylindrical lithium shield is not equidistant from all points on the surface of the cavity. Therefore the flux of γ-rays coming from area A is depressed less by the presence of the lithium than are γ-rays from other parts of the cavity surface. This geometry difficulty tends to give a value for γ_c that is too low. However, because of the rapid fall-off in source strength along the axis of the column, most of the γ-rays in the cavity come from the hemispherical end. For this reason, the error involved is not thought to be prohibitively large.

In summary, then, a reading of 202 "r"/min obtained by exposing the ion chamber in the thermal column without Li shielding may be divided as follows: acquired drift, 20 (10% of total reading), inherent gamma flux, 49 (24%), and thermal neutrons, 133 (66%). Since the thermal flux was 1.80×10^{10} n/cm^2/sec, the number of thermal neutrons required to give a reading of 1 r on a 100-r chamber was $(1.80 \times 10^{10} \times 60)/133 = 8.1 \times 10^9$.

At various times, five different Victoreen 100-r chambers were used. Little difference was noted between the five as far as response to thermal neutrons was concerned.

Absorption of γ-rays by bismuth shell. Conclusions reached up to now refer to the empty graphite cavity. Actually (Fig. 9), the mice were exposed inside a double-layered bismuth shell. The shell consisted of two lids plus two bullet-shaped sheaths. Its purpose was to increase the thermal neutron to inherent γ-ray dose ratio.

Three measurements were made with a 100-r chamber and a solid Li bullet (3.07-cm wall) to determine the effect of the Bi shell on the mixed radiation in the cavity. Results were: no Bi shell —gammas, 22.7 r/min, and thermal flux, 1.96×10^{10} nv (In foils); $\frac{1}{2}$-in. shell—gammas, 16.3, and thermal flux, 1.94×10^{10}; 1-in. shell—gammas, 11.7, and thermal flux, 1.93×10^{10}. Thus, the shell decreases the inherent gamma dose in the cavity by $11.7/22.7 = 0.52$ and has no significant effect on the thermal-neutron flux. In fact, it might have been worthwhile to have used a thicker Bi shield. The neutron fluxes were a little higher than usual for this day's run because of an operational change in the reactor sphere. The

FIG. 17. Bullet-shaped Li shield on graphite wedges, 100-r chamber, and Lucite, Al, Bi and Pb ion-chamber caps

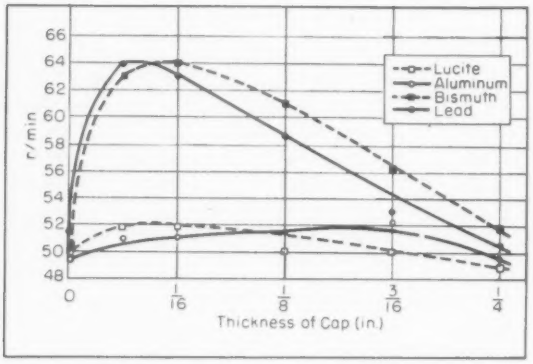

FIG. 18. Effect of ion-chamber caps on 100-r chamber

gamma dose rates observed were, of course, lower than the true total gamma because of the large amount of lithium present.

In good geometry, transmission of a 4-Mev γ-ray by 1 in. of Bi would be $I/I_0 = e^{-\mu x} = e^{-0.415(2.54)} = 0.35$. The actual transmission is a consequence of the spherical bad geometry.

If the inherent gamma dose rate in the empty cavity is 48.7 r/min and if this is reduced by a factor of 0.52 in going through the Bi shells, then the dose received by the mice is (0.52) (48.7) = 25.3 r/min.

Rather than accept this calculated result, the poor geometry Li absorption curve was repeated while the Bi shells were in place. The drift acquired by the 100-r chamber when exposed inside the Bi is the same as without Bi. The first exposure in the Bi shield was made with no Li in place; the reading was taken $5\frac{1}{2}$ min after the end of irradiation. The drift correction was, therefore, 7 r out of the total of 175. The data, after correction for drift, are plotted as curve *A* of Fig. 16. When these data and the data for good geometry (Fig. 16) are normalized to 2.77 cm of Li, curve *B* of Fig. 16 is obtained. Extrapolation of *B* to zero thickness of Li gives an inherent gamma dose rate of 24.3 r/min, to which the mice were subjected.

Subtraction of curve *B* from *A* in Fig. 16 resolves the soft component into a straight line, indicating a slow-neutron flux having an absorption coefficient $\sigma = 66$ barns and an effective average energy of 0.025 ev. This straight line again represents the absorption of thermal neutrons in the Li shield.

Further studies in poor geometry. Further data on the inherent gamma flux of the column were obtained by exposing the 100-r chamber inside the bullet-shaped Li shield without the Bi shells and with caps of Pb, Bi, Al, and Lucite placed over the sensitive portion of the chamber (Fig. 17).

Since the inherent gamma flux has penetrated considerable graphite and Li before reaching the chamber, a secondary-electron equilibrium may be assumed. However, when a material of higher atomic number is added, in the form of a Pb or Bi cap, a new secondary-electron equilibrium must be established at a higher level. If the wall thickness is increased beyond this point of optimum ionization, the primary photon flux is attenuated and there is an over-all decrease in ionization within the chamber.

The curves of Fig. 18 for caps of various thicknesses identify the radiation qualitatively as a γ-ray. However, the complex situation precludes a simple quantitative interpretation.

The curve for Lucite shows no significant increase, thus verifying that equilibrium existed before the Lucite was added. The small maximum in the curve at $\frac{1}{16}$-in. thickness could be due to the slight decrease in electron-stopping power in going from Li to Lucite.

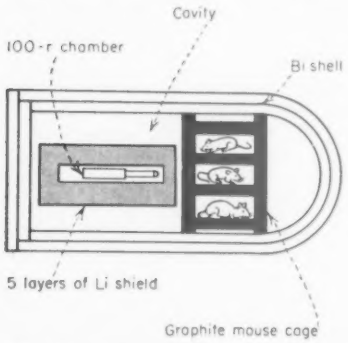

FIG. 19. Arrangement for measuring intermouse gamma effect

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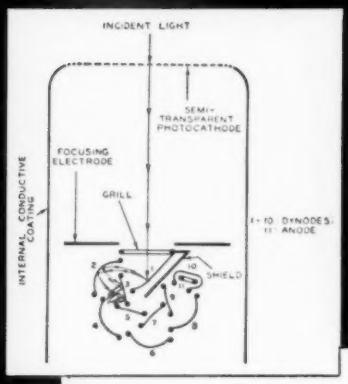
PART TWO OF THIS ARTICLE . . .

appearing in March, discusses the irradiation of mice in the thermal column.

Data are presented from which are estimated the remaining two correction factors necessary for determining the biological effectiveness of thermal neutrons—the intermouse gamma dose and the attenuation effect of the group of mice on the neutron flux.

Results obtained with a 250-kvp X-ray machine are also compared with thermal-column data.

New Multiplier Phototube...



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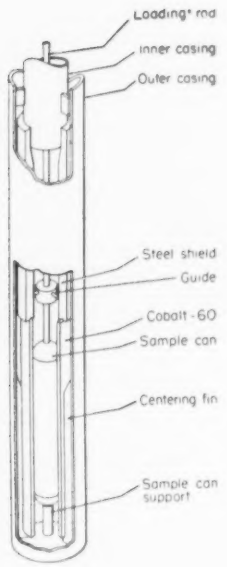

FIG. 1. Source assembly

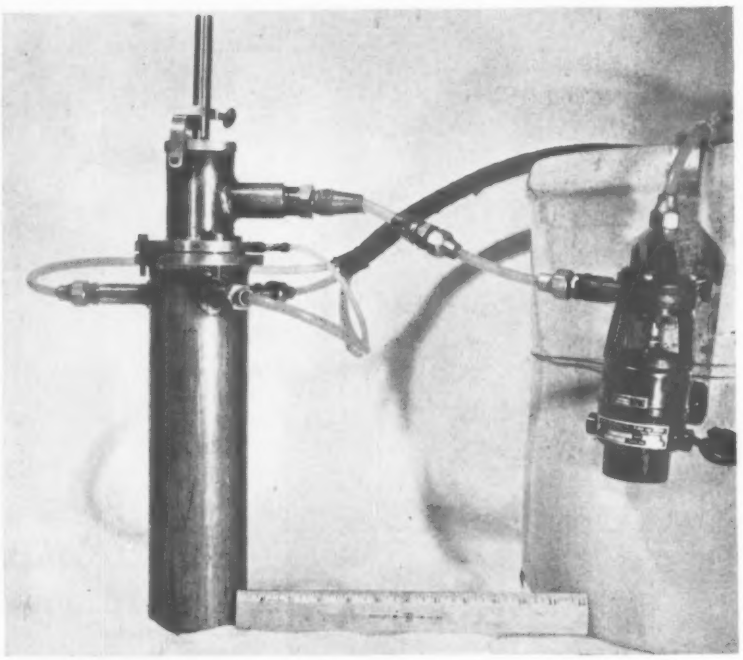

FIG. 2. Top of assembly visible in laboratory

An Economical Water- and Earth-Shielded Cobalt-60 Gamma-Ray Source

By HAROLD A. SCHWARZ and AUGUSTINE O. ALLEN
Chemistry Department, Brookhaven National Laboratory
Upton, New York

Many multicurie Co^{60} gamma-ray sources are currently in use for radiation chemical studies. Most of these require massive lead shielding (*1–4*). The cost of such shielding may become large compared to the cost of the active cobalt, especially when elaborate mechanical arrangements are provided to introduce samples without allowing escape of radiation, and to increase flexibility in the use of the source. In another arrangement (*5*), the cobalt is kept in a pit when not in use, and is drawn to the surface for irradiations. The whole room must then be shielded.

In the present design, the cobalt is kept under water in a pipe buried in the ground under the laboratory floor. Samples for irradiation are placed in a water-tight can that is lowered into the hole. Water and earth are the only shield, and the arrangement is convenient as well as economical. By circulating the water from a thermostat in the laboratory, accurate temperature control is easily secured.

Equipment

The 120-curie source is a tube of cobalt, 12.5 in. long and 1.8 in. inside diameter, encased in anodized aluminum. It was made for us by B. Manowitz and Otto Kuhl, and activated in the Brookhaven reactor.

Figure 1 shows the source assembly. The cobalt rests on three fin-like supports welded to the outer container, which is a stainless-steel tube 3 in. ID and 7½ ft long, buried under the laboratory floor to a depth of 6 ft. A stainless-steel inner tube rests on the source; the bottom 12 in. of this tube is made ½-in. thick to provide extra shielding.

Water from a thermostat is pumped at about 1 gal/min down the inner tube and emerges from three outlets near the top of the outer tube. The top is provided with water-tight fittings since the inlet water is under pressure. Figure 2 shows the top of the assembly.

The aluminum sample can is attached to the end of a long stainless-steel rod, jointed in the middle to allow bending when pulled out. A tapered prong on the end of the can fits into a hole at the bottom of the container; this, together with a three-pronged guide on the rod, provides accurate positioning. When samples are being inserted or removed, the water

flow is turned off and the top closure removed.

The source was brought to the laboratory in a lead container fitted with a heavy lead door. The container was suspended over the pipe from a crane and the door opened, allowing the source to drop into the pipe without significant exposure of personnel to radiation.

Performance

The shielding is overdesigned; the radiation level at the surface is undetectable. By raising the source to various levels under water (without the inner tube present) and extrapolating to full depth, we find a level of 0.07 mr/hr at the most intense spot; presence of the inner tube with its heavy lower part should reduce the radiation escaping by another factor of 10, to 0.007 mr/hr. The present design would apparently offer completely adequate shielding for a 12,000-curie source. In making another assembly of this kind, we could omit the heavy stainless-steel collar on the end of the inner tube, and reduce the depth of the hole by a foot for a source of the present strength.

The intensity within the source in March, 1953, was 47,800 r/hr (based on $FeSO_4$ dosimetry (*6*, *7*), taking 15.8 molecules $FeSO_4$ oxidized per 100 ev and 93 ergs per gram-roentgen) and was reproducible and uniform to ±1%. The temperature is usually kept at 25° C, and the difference in temperature between inlet and outlet water is less than 0.1°. When the thermostat temperature was raised to 65°, the temperature difference between inlet and outlet water fell to 2° within three hours.

The source has been in frequent use for several months and has proved convenient and trouble-free. The assembly is highly recommended as an economical source for radiation chemical work.

* * *

This research was carried out under the auspices of the Atomic Energy Commission.

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TABLE 1—Uranium Contaminated Glass Surfaces

Solution of $UO_2(NO_3)_2$				*Instrument response**			
Plate	*U-concentrate (mg-U/cm³)*	*Volume (cm³)*	*contamination (mg-U/cm²)*	*(mrep/hr/mg-U/cm²) 11*	*12*	*13*	*14*
I	198	0.822	1.0093	1.100	0.842	0.803	0.733
II	4.17	1.020	0.0264	1.515	1.515	1.515	1.515
III	417	0.760	1.9647	0.840	0.713	0.733	1.008
IV	417	0.720	1.8611	0.860	0.752	0.752	1.048
V	19.8	0.830	0.1019	1.079	1.079	1.079	0.981
			Average	0.900 ± 0.148 S.D.			

* Instruments used were: 11, Nuclear-Chicago model 2610, serial no. 542; 12, Nuclear-Chicago 2610A, no. 553; 13, Nuclear-Chicago 2610A, no. 201; and 14, Victoreen 263A, no. 302.

Estimating Radium and Uranium Surface Contamination

By EUGENE V. BARRY, LEONARD R. SOLON and HANSON BLATZ

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U. S. Atomic Energy Commission
New York, New York

An increasingly important problem, in plants processing uranium or radium bearing materials, is the evaluation of the mass of radioactive substance on a surface, in order that the significance of this contamination can be judged. In such situations as the release of contaminated steel into unrestricted scrap-metal channels or the return of a plant leased by AEC, for a term of years, to the owner, it is desirable that a reasonably reliable index of actual mass contamination be obtained.

An attempt has been made along this line in the present study. Known amounts of radium and uranium were used to contaminate steel and glass plates, and their subsequent activity measured with nuclear survey instruments. Indices have been obtained relating the measured activity to the specific contamination measured in mass of radioactive material per unit area.

The limit of uranium contamination one may reasonably hope to detect when radium contamination is also present is discussed.

Uranium Contamination

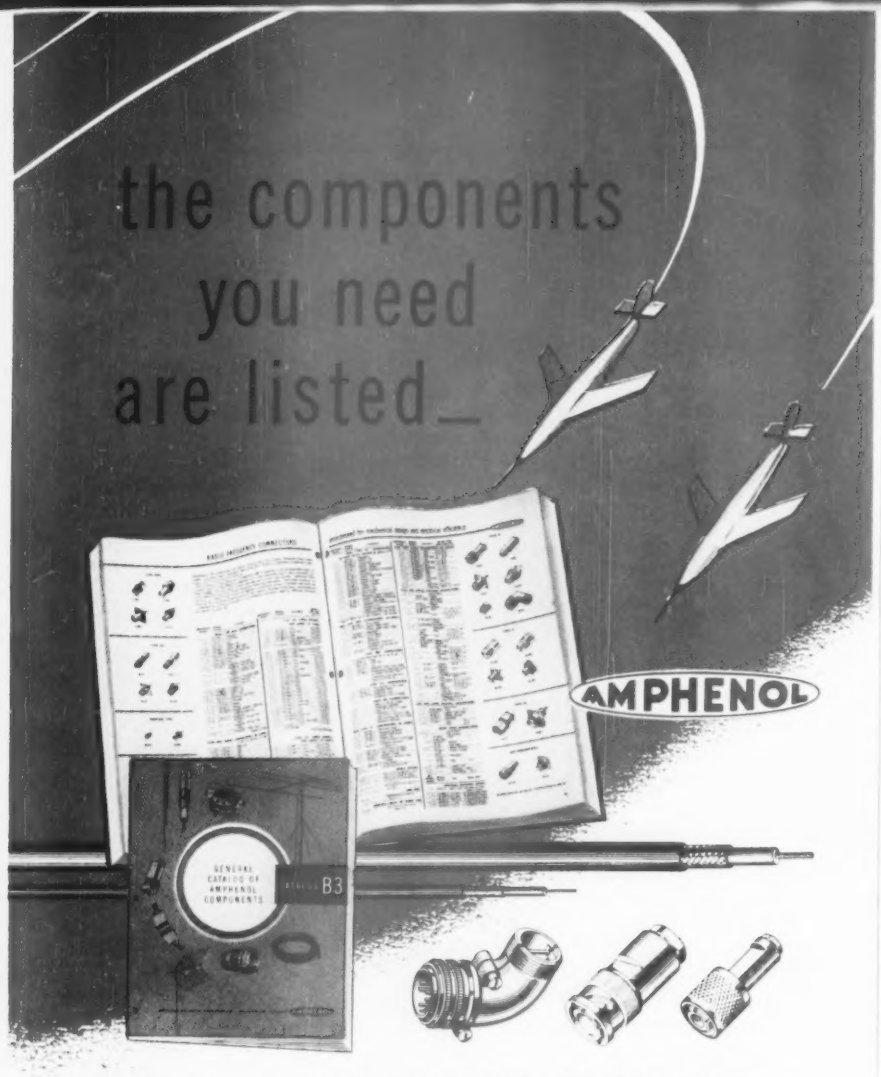
Solutions of uranyl nitrate [$UO_2(NO_3)_2\ 6H_2O$] having different concentrations were prepared. The solutions were of sufficient age to permit the U^{238} to have achieved equilibrium with UX1 and UX2.

From each solution 25 drops were transferred from a Mohr pipette more or less uniformly and centrally to each square inch of a 5 × 5-in. glass plate. The total deposition was recorded, the average mass of uranium per unit area calculated, and the response of representative G-M survey instruments observed with the shield open and with the side of the probe held in contact with the plate. At this level, the slight gamma activity (shield closed) from uranium is not detectable with this type of instrument.

Three instruments were used, all calibrated using Co^{60} standards. Results of the measurements are exhibited in Table 1.

For survey work using these instruments, an approximate rule of thumb is that a meter reading of 1 mr/hr*

* Values of mr/hr with the shield off are merely meter readings and not actual beta dose rates, which are about twice the meter readings.



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with the shield off represents approximately 1 mg of natural uranium in equilibrium with UX1 and UX2 per cm² of surface. The actual index obtained for uranium contamination was 0.90 ± 0.15 mr/hr per mg of U/cm².

In this calculation the results obtained with Plate II were neglected since the readings were only slightly above background, i.e., 0.04 mrep/hr. Because of the normal fluctuations in radiation count experienced with these G-M instruments, the evaluation of a contamination level lower than 0.02 mg-U/cm² would be necessarily a very rough approximation.

Radium Contamination

Using a similar procedure, a radium-salt solution was used to contaminate a variety of 5 × 5-in. steel and glass plates. Disposition of the plates is shown in Table 2. Measurements with the shield open and closed of the subsequent activity were made with an AEC class-SGM survey meter. The results are exhibited in the figure on page 61 and Tables 2 and 3.

The figure shows the buildup of the gamma activity with time normalized to 0.01 μg Ra/cm². Analysis of the data obtained with groups *A* and *B* indicated that the gamma response (shield closed) of the detector was independent of the type of surface contaminated, provided the surface was covered with rubber hydrochloride. For this reason the covered-plate data have been grouped together in the figure.

The difference in the average gamma response, after six or more weeks, between the covered and uncovered steel plates was small, but statistically significant at the 0.02–0.01 level according to the Student *t* test.

Group *D*, the exposed and weathered steel plates, experienced an approach to equilibrium but then proceeded to show diminished activities as the contamination was presumably removed by rain and wind. The ratios of the open-window to shielded instrument responses, designated here as *O*/*S*, are exhibited in Table 3. The data could not be grouped in the same manner as in Table 2; the difference in the average open-window response between groups *A* and *B*, the covered glass and steel plates respectively, although not large, were highly significant (< 0.001 level) using the Student *t* criterion.

Also significant, between the 0.05–

TABLE 2—Gamma Response of Radium-Contaminated Surfaces

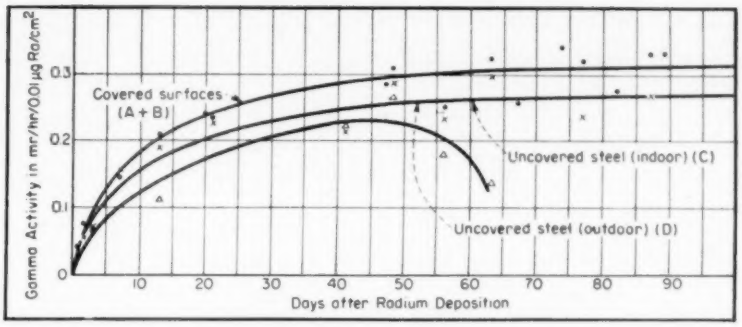
Days after deposition	*Response (mr/hr/0.01 μg-Ra/cm²)**		
	A + B	*C*	*D*
1	0.038	0.021	0.024
2	0.078	0.055	
3	0.067	0.072	
7	0.145	0.154	
13	0.208	0.190	0.110
20	0.239		
21	0.235	0.226	
44	0.226	0.217	0.226
47	0.288		
48	0.307	0.289	0.271
56	0.253	0.235	0.181
63	0.325	0.298	0.138
67	0.258		
74	0.344		
77	0.326	0.235	
82	0.279		
87	0.334	0.271	
89	0.335		
103	0.309		
113	0.322		

* **Group A:** glass plates 1, 2, 3, 4; each contaminated with 2.35 micrograms of radium (0.0146 μg/cm²) and covered with 0.0005-in. rubber hydrochloride. Kept indoors.

Group B: steel plates 1, 3; each contaminated with 2.23 micrograms of radium (0.0138 μg/cm²) and covered with 0.0005-in. rubber hydrochloride. Kept indoors.

Group C: steel plates 2, 4; each contaminated with 2.23 micrograms of radium and left uncovered. Kept indoors.

Group D: steel plates 5, 6; each contaminated with 2.23 micrograms of radium, left uncovered, and exposed outdoors on the roof of the New York Operations Office.



GAMMA RESPONSE of various contaminated surfaces

TABLE 3—Open-Window Gamma Response (*O/S*) of Radium-Contaminated Surfaces

Days after deposition	*Group A*	*Days after deposition*	*Group B*	*Days after deposition*	*Group C*	*Days after deposition*	*Group D*
1	93.3	1	146.4	1	174.2	1	153.0
20	46.5	2	73.2	2	101.0	13	63.0
47	40.8	3	102.8	3	97.6	41	48.7
67	47.5	7	66.9	7	63.9	48	42.5
74	38.8	13	48.7	13	54.4	56	61.0
82	40.7	21	52.3	21	51.0	63	67.6
89	37.9	41	52.9	41	54.2	77	
103	41.6	48	46.6	48	48.6		
113	40.4	56	48.4	56	51.4		
		63	42.0	63	41.3		
		77	44.3	77	57.9		
		87	43.0	87	52.3		

0.02 level, was the difference in the average response of the covered and uncovered steel plates, Group *B* and *C*.

Summary

The numerical results of the experiment are summarized here for the equilibrium conditions prevailing after seven weeks or more. Group *D*, the exposed plates, are not included, since the activities are subject to fluctuations of weather, and true radioactive equilibrium would probably never be achieved.

Covered plates. Gamma response was 0.315 ± 0.040 mr/hr/0.01 μg Ra/cm². With covered glass plates, the ratio of unshielded to shielded reading was $O/S = 39.9 \pm 3.5$. With covered steel plates the ratio was $O/S = 44.8 \pm 3.2$.

Uncovered plates. Gamma response was 0.266 ± 0.034 mr/hr/0.01 μg Ra/cm². With uncovered steel plates the ratio was $O/S = 50.3 \pm 6.6$.

Conclusions

The following criteria for surface contamination are indicated for the type of survey instruments used in the experiment. For uranium of natural isotopic composition in equilibrium with decay daughters UX1 and UX2, 0.9 mr/hr per 1 mg−U/cm² (gamma shield open).

For radium contaminated surfaces in which the radium bearing material has been essentially undisturbed for six or more weeks, 0.3 mr/hr per 0.01 μg-Ra/cm² (gamma shield in place), and 12–15 mr/hr per 0.01 μg-Ra/cm² (gamma shield open).

Assuming it is feasible to read a survey meter by an amount 0.03 mr/hr above normal background, one could hope to detect 30 μg/cm² of uranium on a contaminated surface.

However, it should be emphasized that as little as 2×10^{-5} μg/cm² of radium will yield essentially the same response with gamma shield removed. The fact that the open-window to shield response for radium is about 45 to 1 must be borne in mind always by the careful surveyor when examining surfaces where mixed uranium and radium contamination is possible.

A good working rule is this—a reading of at least 1.5 mr/hr, with the G-M tube gamma shield removed, shall be taken as evidence of uranium contamination if the meter response through the gamma shield reads background on contact with the surface. If there is no evidence one way or another, a reading of less than 1 mr/hr with the shield removed is ambiguous and may arise from either radium (less than 0.001 μg/cm²) or uranium (less than 1 mg/cm²) contamination, or an equivalent combination of both. The easy temptation for the surveyor to infer "beta," and thus uranium, contamination should be avoided.

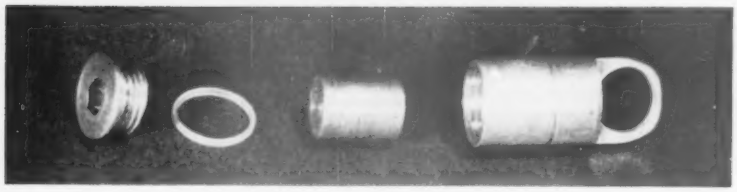


FIG. 1. Disassembled Po-Be source

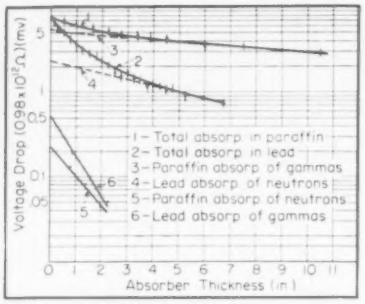

Voltage Drop (0.98×10¹²Ω)(mv)
5
1
0.5
0.1
0.05
1 – Total absorp. in paraffin
2 – Total absorp. in lead
3 – Paraffin absorp. of gammas
4 – Lead absorp. of neutrons
5 – Paraffin absorp. of neutrons
6 – Lead absorp. of gammas
0 1 2 3 4 5 6 7 8 9 10 11
Absorber Thickness (in.)

FIG. 2. Absorption curves for Po-Be source no. 186, with brass-wall ionization chamber

Gamma Dose Rate from a Po-Be Source

By GERALD J. RAUSA
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University of Rochester
School of Medicine and Dentistry
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In an experiment involving the exposure of X-ray film to a polonium-beryllium source, it was desired to know the gamma dose rate emitted from the source that would contribute to the density of the film. The present experiment was carried out to estimate this gamma dose rate.

The gammas from a Po-Be source are attributed to the de-excitation of the excited levels of the residual C^{12} nucleus (4.4 Mev primarily), the gamma emission from Po^{210} (0.77 Mev), and others that may arise from any inelastic collisions of neutrons with the container material, or as a result of the activation of the holder by neutrons.

Absorption curves were made using a $\frac{1}{16}$-in. brass-wall ionization chamber with outside dimensions of $2\frac{5}{8}$-in. diameter and 3-in. height as the receptor. This was used to drive a vibrating-reed electrometer measuring the voltage drop across a 0.98×10^{12} ohm resistor, rather than using a rate-of-drift method. The data were recorded by means of a Brown Elektronik potentiometer. The electrometer head showed no response to extra-cameral radiation as used in the experiment.

The source was contained in a right cylinder of 0.33-in. diameter and 0.460-in. height (outside dimensions) with a wall thickness of 0.02 in. of nickel. This was coated with nickel to 0.37-in. diameter and 0.500-in. height. An outer container was made of aluminum, with external dimensions of 0.5-in. diameter and 0.75-in. height. Figure 1 shows an exploded view of the source. The neutron emission on July 6, 1953 was 0.202×10^7 n/sec. The source was placed at 60 cm from the center of the ionization chamber for the experiment.

The absorbers were $\frac{1}{4}$-in. thick by 3-in. diameter lead, and 3-in. diameter paraffin slabs of varying thickness.

The absorption curve, as shown in Fig. 2, plots the voltage drop across the resistance as a function of absorber thickness. Since it was known that the highest gamma energy is 4.4 Mev (*1*), this was used as a starting point. The absorption coefficient for 4.4 Mev gamma in paraffin is 0.024 cm^{-1} (*2*), or a half-value thickness of 11.37 in. The best tangent to the paraffin absorption curve having this slope was then drawn. The intercept with the origin is 5.4 mv. Since paraffin absorbs neutrons more readily than gamma, the 5.4 mv is a measure of the gamma dose. The difference between the maximum 7.7 mv and 5.4 mv must then be a measure of the response of the instrument to neutrons. This value, 2.3 mv, was plotted on the ordinate at zero absorber and the best tangent to the lead absorption curve then drawn. The slope of this line is indicative of the absorption coefficient of lead for Po-Be neutrons. This value is found to be 0.0642 cm^{-1}.

The plot of the differences between the absorption curves and the tangents to the curves is a measure of the absorption coefficient for neutrons in paraffin and the 4.4-Mev gamma in lead. These two figures were calculated to be 0.321 cm^{-1} and 0.436 cm^{-1} respectively (from the curve). The absorption coefficient in lead for 4.4-Mev gamma has been computed to be 0.463 cm^{-1} (*2*), which seems to indicate that although the method is crude, the results are reasonably accurate. The 0.77-Mev gamma does not seem to show except for the fact that the absorption coefficient of the 4.4-Mev gamma is not quite correct. The possibility of some gamma arising from inelastic collisions has not been taken into account either.

Another method of obtaining the gamma dose rate, a method that is somewhat independent of the energies of the gammas emitted, would involve the use of scintillation counters with air-equivalent crystals. But the crystals are also neutron sensitive.

It is of interest to calculate the neutron sensitivity of the ionization chamber. The neutron flux at the center of the chamber was 44.6 n/cm^2/sec. This gave rise to a current of 2.35×10^{-15} coulomb/sec, in which case the response of the instrument is 5.27×10^{-17} coulomb/n/cm^2.

The gamma dose rate is calculated as

$$\frac{5.4 \times 10^{-3}}{0.98 \times 10^{12}} \times \frac{3{,}600}{288} \times 3 \times 10^9 = 0.2066 \text{ mr/hr at 60 cm}$$

where 5.4×10^{-3} = voltage drop, 0.98×10^{12} = bleeder resistance, 288 cm^3 = chamber volume, 3×10^9 = esu/coulomb, and 3,600 = sec/hr.

With this type of source, therefore, the ratio of the gamma dose rate in mr/hr to neutron flux in n/cm^2/sec is 0.00463.

The suggested maximum permissible exposure (MPE) to Po-Be neutrons is 34 n/cm^2/sec and that to gamma radiation is 7.5 mr/hr. It can be computed, therefore, that at a distance from the source that will give the tolerance dose for neutrons, the gamma dose rate is only 2.1% of the suggested MPE.

* * *

This article is based on studies performed for the Atomic Energy Commission under contract W-7401-eng-49.

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2. "Health Physics Handbook" (Health Instrument Division, Hanford Works, General Electric Co., Richland, Wash., 1953)

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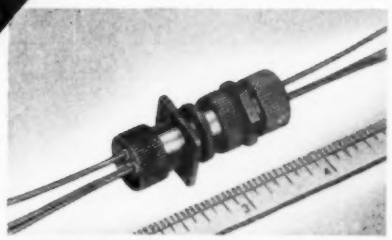

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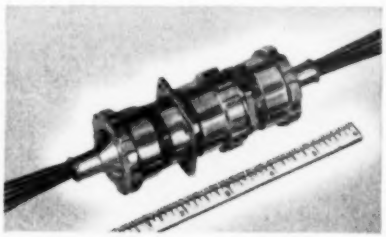

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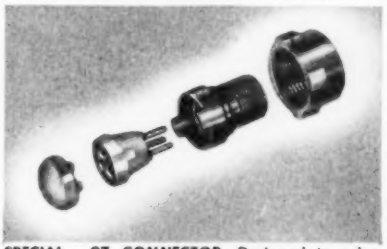

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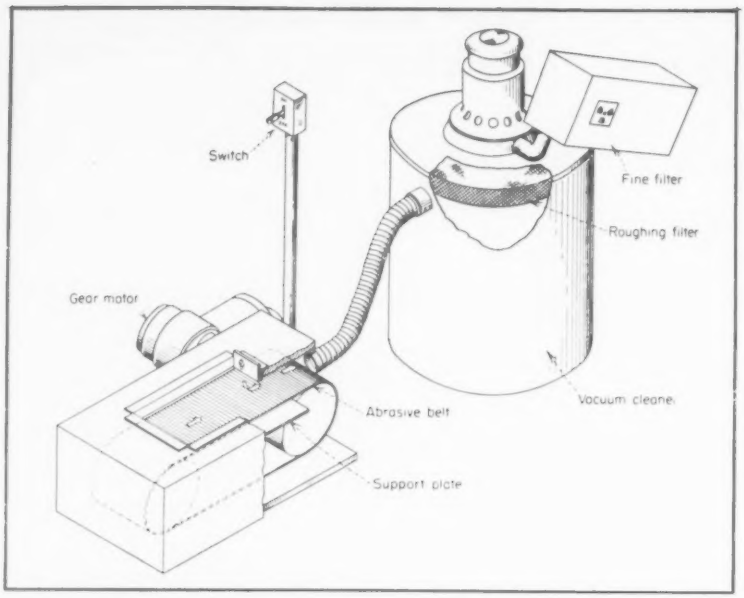

SHOE SANDER removes contamination from soles quickly, is made from parts of obsolete laboratory equipment. Picture below shows unit in use

Mechanical Shoe Sander

By JERE C. AUSTIN
Hot Laboratory Operations Division
Brookhaven National Laboratory
Upton, New York

The illustration on this page shows a practical and efficient method of adapting an ordinary belt sanding machine for decontaminating shoes. Its components are parts of obsolete laboratory equipment.

The sander itself is enclosed in a box made of 1-in. plywood. The hole in the top was cut as wide as the belt and long enough to accommodate the largest anticipated shoe size. A metal strip was fastened around the shoe opening to prevent wear of the plywood and to take up clearance between the sander bed and the box cover. A half-inch clearance was left at the front end to provide a path for air and contaminated dust to the vacuum cleaner.

The machine runs at a slower speed than for ordinary sanding; in this case 86 rpm, provided by a $\frac{1}{6}$-hp motor and reduction gear, is sufficient.

The vacuum cleaner was modified by fastening a glass-wool roughing filter with wire mesh at the blower inlet and a commercial 'Absolute'* filter at the outlet.

Sander and vacuum cleaner run simultaneously from a spring-loaded switch that automatically return to "off" when pressure is released.

The unit has been in use approximately six months. It has not yet been necessary to renew the sanding belt. The belt has been contaminated twice—enough to transfer contamination to other shoes. In one instance, it was decontaminated by holding a

* Cambridge Corporation, Syracuse, N. Y.

stiff brush against the belt while the unit was running. Another time, the belt was scrubbed with detergent and allowed to dry.

The use of this unit can result in a general increase in efficiency by cutting down on time consumed by personnel in decontaminating foot gear.

* * *

This work was done under the auspices of the Atomic Energy Commission.

Labeling Bovine and Human Albumin with I^{131}

By RICHARD C. GILMORE, Jr., MARGARET C. ROBBINS and ALLEN F. REID
Department of Biophysics
Southwestern Medical School
University of Texas, Dallas, Texas

Use of I^{131} for the labeling of proteins and related compounds has been adapted to a large extent because the isotopes are available, and most important, because the present methods for the labeling of these compounds have proved ineffectual in altering their "physiological" structure. We became interested in labeling albumins with I^{131} in an attempt to increase the yield of labeled product over that reported by the methods appearing in the current literature. Yields of more than 90% are obtained, and no detectable alterations in the protein molecule have been noted.

Development of Techniques

The method used for labeling proteins with iodine in the present investigation is basically the same as that which appears in the literature and has been employed since 1876 (*1*) when it was first reported that iodine combines with protein. Although the different methods vary in reagents used and procedures employed, all are basically the same in that iodide is oxidized to elemental iodine, which then substitutes onto the protein molecule to give the iodinated product. Various oxidizing agents have been used, and the protein is placed in either a buffered or an alkaline solu-

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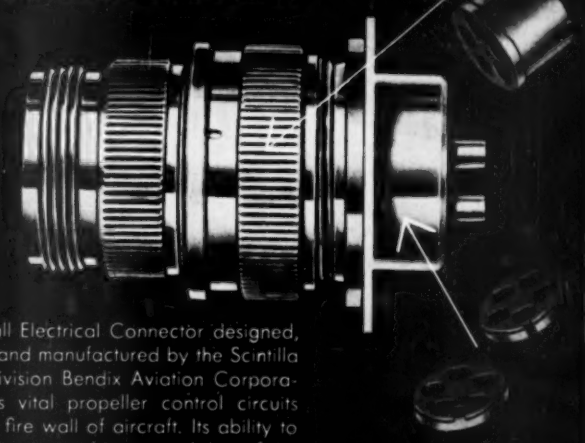

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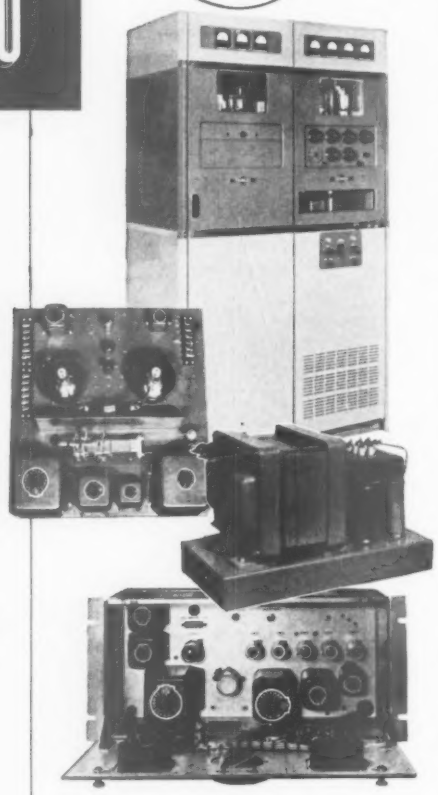

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tion during the substitution reaction.

Earlier methods invariably produced undesirable alterations in the protein molecules as evidenced by oxidation of sulfur (*2*) and by changes in the color reaction of the proteins to Millions and Adamkiewiez reagents (*2, 3*) indicating an alteration in the tyrosine and tryptophane residues of the protein molecule, respectively. It was the work of Drechsel (*4*) in isolating iodogorgonic acid and the efforts of Wheeler and Jamieson (*5*) and Henze (*6*) in identifying it as diiodotyrosine that drew attention to tyrosine as the group responsible for the iodine-binding properties of protein molecules; this initiated the work that eventually eliminated the detectable alterations of proteins during the iodination procedure. Final proof that tyrosine was the compound in the protein molecule to which iodine was bound has been credited to Oswald (*7, 8, 9*).

From the known tyrosine content of various proteins it was observed by these early workers that more iodine was added to these compounds than could be accounted for by their tyrosine content. Pauly (*10*) demonstrated that iodine would also add to the imidazole group of histidine. However, this reaction occurs only if the amount of iodine present in the reaction medium exceeds that amount capable of being substituted onto the tyrosine moieties of the protein molecule.

Blum and Vauble (*11*) first introduced a bicarbonate solution to buffer the hydriodic acid formed during the iodination of proteins. Later, Blum and Strauss (*12*) iodinated proteins in a solution containing ammonium hydroxide and believed this permitted iodination with less alteration of the protein molecule than did the method that made use of the bicarbonate solution. Wormall (*13*) iodinated sera from various species by a modification of the method of Blum and Strauss.

The use of immunological methods for studying the effects of iodination on proteins was very helpful as slight alterations in the structure of serum proteins will produce detectable changes in the specificity of antibodies produced by these sera. Wormall observed that the serum proteins gave negative Millions tests and that antibodies produced by the halogenated sera were different from those produced by unhalogenated sera—the former almost completely having lost their species specificity. He did ob-

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serve, however, that the incomplete iodination of proteins permits some species specificity of the sera to remain and does not obliterate the Millions test, which indicated less alteration in the protein molecule than heretofore reported.

This then led to subsequent work in an effort to determine the optimum amount of iodine that could be substituted onto a protein without altering its immunological structure. Breine and Haurowitz (*14*) studied the action of iodine on agglutinating antisera and reported that more than 1% incorporation of iodine into the protein resulted in a marked reduction in precipitable power. Although the method of Fine and Seligman (*15*) was first used to provide radio-labeled protein for studies of alterations in plasma volume during shock, other investigators have used their procedure for labeling proteins with I^{131} for immunological studies with no apparent change in the structure of the protein. Using rabbit antisera globulin, Butement (*16*) calculated that the substitution of more than 10 atoms of iodine per molecule of protein produced a marked decrease of precipitable activity; from an empirical formula he calculated that 7.7 atoms of iodine per molecule of this globulin was the optimum ratio to obtain and still retain full antisera activity. These observations are in agreement with the work of Francis, Mulligan and Wormall (*17*), who encountered no detectable decrease in antibody properties when their method of "trace-labeling" was used. This resulted in a ratio of 13 atoms or 6.5 molecules of diiodotyrosine per molecule of antibody, assuming a molecular weight of 165,000 for the globulin they employed. When using I^{131} that is available at the present time, it may be assumed that the solution is "carrier-free," and the optimum amount of iodide can be calculated and added to the reaction mixture as a carrier for the radioisotope.

Present Procedure

While all of the preceding methods will result in a product that is apparently unaltered, the yields vary from fractions to about 30% based on the amount of I^{131} used. The procedure reported here gives yields of labeled protein consistently better than 90%, based on the I^{131} used, and no noticeable alterations have been produced in the protein molecule although immunological or electrophoretic studies

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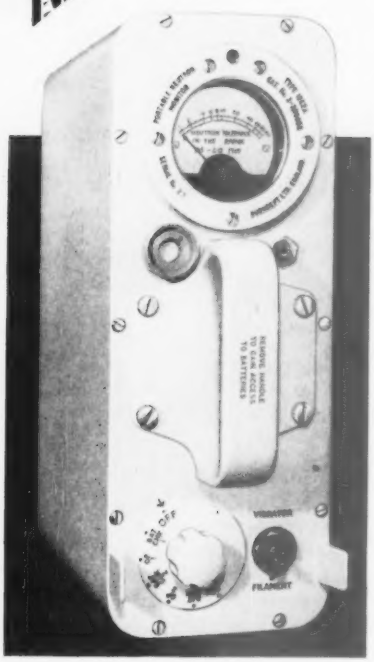

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have not been carried out to determine this accurately. Labeled bovine albumin prepared by this method has been used with very successful results (*18*), and no pyrogenic reactions have been observed. Assuming the I^{131} to be "carrier-free" and albumin as having a molecular weight of 70,000, the ratio of total iodine molecules used in each experiment to the number of molecules of albumin used calculates to be 3:1, and is in excellent agreement with that reported in the literature of minimizing alterations in the protein molecule during the labeling procedure, even if all of the iodine were incorporated into the protein.

Experimental

The following reagents were used: 1. Phosphate-saline buffer (8.46 gm NaCl, 7.24 gm $Na_2HPO_4{\cdot}12H_2O$, 0.53 gm $NaH_2PO_4{\cdot}H_2O$), made to volume of 1 liter with distilled water. 2. 2.5% (w/v) guanidine hydrochloride solution (Eastman 749). 3. 0.217% (w/v) potassium iodide solution. 4. 10.0% (w/v) ammonium persulfate made up in the phosphate-saline buffer. 5. Bovine albumin powder, fraction V from bovine plasma (Armour Laboratories, Inc., Armour and Company, Chicago, Ill.). 6. U. S. P. salt-poor solution of normal human serum albumin (Code 684-20, Cutter Laboratories, Berkeley, Cal.). Solution evaporated to dryness and the solid material used.

Procedure. The following procedure is used.

1. Dissolve 150 mg of bovine albumin in 2 ml of phosphate-saline buffer contained in a 35-ml test tube.

2. Add 2 ml of the guanidine hydrochloride solution and mix.

3. Add 1 ml of potassium iodide solution and mix well.

4. Add the I^{131} and mix. Usually 10 to 20 mc in 0.5 to 2 ml volume has been employed, although 64 mc in the least possible volume was used successfully in one experiment.

5. Add 2 ml of ammonium persulfate solution slowly (one minute) with gentle mixing of the contents of the test tube during the addition.

6. Let the solution stand in the hood for 4 hr, then transfer the contents of the test tube as quantitatively as possible to 1-in. Visking cellophane dialysing tubing and dialyze against tap water for 2 hr. Then transfer the tubing and contents to a 400-ml beaker containing 300 ml of phosphate-saline buffer and let stand for ½ hr. The labeled albumin is ready for use. The dialyzing tubing has not been treated to remove the possible pyrogens present, but no such reaction has been observed during the time that the labeled albumin has been used in animal experimentation.

The solutions, with the exception of the phosphate-saline buffer, are prepared just before they are needed for each experiment. Comparable results have been obtained when human albumin is used in place of bovine albumin. In one experiment, the yield of I^{131} labeled bovine albumin was not appreciably different when 500 mg of bovine albumin in 2 ml of phosphate-saline buffer was employed, no other changes being made in the procedure outlined.

* * *

This investigation was supported in part by research grant H-1261 (C-2) from the National Heart Institute of the National Institutes of Health, U. S. Public Health Service. The radioisotope was furnished upon allocation by the U. S. Atomic Energy Commission.

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TECHNICAL BRIEFS

• **Stability of binary scaling** circuit is improved by use of SiC nonlinear resistors in the plate-to-grid coupling circuits. Speed of scaler is increased for the same tube life expectancy, according to J. L. Lerner in ANL-5072, because larger fraction of plate-voltage swing can be applied to the opposite grid.

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BOOKS

Experimental Nuclear Physics, Vol. 2

E. SEGRÉ, Editor (John Wiley & Sons, Inc., New York, 1953, viii + 600 pages, $12.00). Reviewed by R. F. TASCHEK, Los Alamos Scientific Laboratory, Los Alamos, N. M.

This second volume of the Segré trilogy contains Part VI, a two hundred page "Survey of Nuclear Reactions" by Phillip Morrison, and Part VII, somewhat less than four hundred pages on "The Neutron" by Bernard Feld.

Nuclear Reactions

Perhaps the simplest characterization which can be given of the article on nuclear reactions is that of "physical picture," which is as it should be in an exposition of theory for a volume on experimental nuclear physics. Morrison has succeeded remarkably well in explaining in words the general and even detailed aspects of nuclear reactions from very low energies into the 100-Mev region. This is not to say that formulae and derivations are minimized, but rather that they are used to point up the argument and understanding of the processes being explained. It would, however, not be generally possible for experimentalists to use the formulae in this volume to calculate fits to data, i.e., the treatment is not in sufficient quantitative detail.

The approach chosen emphasizes primarily the use of compound nucleus formation with subsequent decay, following the Weisskopf school. Thus almost no mention is made of the shell model and of the specific predictions of level ordering for the first few levels, although the dispersion theory is discussed in detail.

Part VI begins with a discussion of the conservation laws, which contains a derivation in relativistic coordinates of the dynamics of nuclear collisions; solid angle transformations might have been added, but experimentalists perhaps need to be left with some derivations. Next comes a qualitative discussion of what methods one uses to get nuclear reaction data and an introduction to the compound nucleus concept. This is followed by a listing of advantages and failings of various nuclear models and by a derivation and calculation for level widths; the barrier penetration factors are derived and some of them graphed in Appendix I. In sections 5 and 6, the statistical model is discussed with great clarity, first neglecting isolated resonances and then in detail for the one-level resonance case including level shift and effects of the coulomb barrier. Typical resonant and nonresonant cross sections as functions of energy are discussed in section 7. Next, two chapters are devoted to the special cases of low-energy deuteron reactions and to gamma-ray deexcitation of excited nuclei, respectively; isomerism is briefly discussed but not specifically as a state of a residual nucleus. The fission process is only briefly discussed in section 10; the general picture is given, but no experimental results.

In the earlier sections, the groundwork was laid for *Nuclear Reactions at High Energy*, which, to the best of the reviewer's knowledge, is a subject not treated elsewhere as a unified extension of the classical energy range of nuclear reactions. This point of view is well carried out here; meson formation is, of course, omitted. The "nuclear cascade" is discussed in terms of the Monte Carlo method for the many intranuclear collision effects; single collisions within the nucleus and the "pick-up" effect are included followed by a statistical treatment of spallation and evaporative stars. Some data is presented, but the greatest usefulness of this section will be in getting an over-all picture of the behavior of the nucleus when large amounts of energy are pumped into it.

Almost no errors or misstatements were found; on p. 22, total cross sections for incident charged particles are not more easily measured than differential cross sections if the energies are below about 30 Mev. Again, on p. 141, one would not really expect the fission cross section for an element with a threshold to be zero at a definite energy. No complete listing of reactions nor even of all types is given, but references to survey literature are quoted. In general, however, Part VI is somewhat sparsely referenced.

The Neutron

Feld's discussion on "The Neutron" has a far more orthodox form and content than the survey of nuclear reactions. The neutron is first described according to the latest data on charge, mass, stability, spin, and magnetic moment, following which the simplest interactions (with protons and electrons) are discussed. It is interesting to find listed here also those experiments that presume to study directly the neutron-neutron interaction.

The two long sections that follow the basic introduction cover, first, *Interaction with Nuclei* and then *Sources and Detectors*. The compound nucleus formulation is generally followed in the discussion for neutron interactions, there being some repetition of subjects in Part VI such as the penetrabilities listed on pp. 51 and 262. The explanation and figures for level spacing, resonance theory and potential scattering are excellent. There are figures of observed total cross sections compared with theory for specific regions of the periodic table. It is unfortunate that the book came out too soon to include the higher energy total cross-section data of the Wisconsin group, which has led to the "cloudy crystal ball" theory; mention of these data is made in a footnote on p. 283. Useful tabulations of slow-neutron scattering cross sections, resonance parameters, radiative-capture cross sections, and thresholds for (γ,n) and $(n,2n)$ reactions are given. A good discussion of the difficulties and inadequacies of inelastic scattering measurements is given, a field in which there is a great deal of activity at present. The other partially unsolved problem of measuring the angular distributions of elastic scattering in the few-Mev range is also well treated.

An excellent summary of the sources of neutrons used for measurements in all energy ranges from the thermal region thru the 100-Mev stripped deuteron and knock-on neutron region is given in section 3. The McKibben nomograph for $H^3(p,n)He^3$ is outdated because of the new threshold at 1,019 kev. The great usefulness of the long counter as a detector might have made it desirable to include the results of Kusherniuk on how to optimize flatness of response in different energy regions.

A short section on the *Interaction of Neutrons with Matter in Bulk* covers slowing down and diffusion of neutrons in a very readable fashion; elementary pile theory is given in a brief form, but contains most everything of greatest interest to the experimentalists using piles as neutron sources.

The book closes with a section on *Coherent Scattering Phenomena*, an active and prolific field which has strong analogies to X-ray diffraction. The experimental techniques, information and theory are clearly presented.

Part VII seems to have a complete set of references to the unclassified literature.

This book is a highly desirable one for any practicing nuclear physicist to have. The authors are to be complimented on being able to bring so near to date the information in this rapidly moving field.

Report on the Atom

By GORDON DEAN (Alfred A. Knopf, Inc., New York, 1953, vii + 321 pages, indexed, $5.00). Reviewed by JEROME D. LUNTZ, Editor, NUCLEONICS.

Whether he likes it or not, atomic energy today affects the life of every person in this world. Consequently, in addition to those working with atomic energy, it behooves laymen to have some feel for the significance of the atom. However, there are some laymen—e.g., the nontechnical commissioners of the Atomic Energy Commission—who must go several steps farther.

This book then is the story of what one commissioner, former chairman Dean, learned when he had to wrestle with the atom in his day-to-day work. It is a fine, readable survey of what this country's atomic energy program is and what some of the problems are. It was intended to be and is, for the most part, an impersonal story. This is a bit unfortunate in that, because AEC must necessarily operate in secrecy, the public has no clear picture of the problems that beset this agency.

There are some places where Mr. Dean gives the reader a glimpse of some of these problems. On the relationship between the Pentagon and the AEC, Mr. Dean says, "The task of getting along with the military . . . is fairly easy, although it is sometimes bothersome." One reason for this is apparently that the military suggests "not simply *what* the Commission should do [on weapons production requirements] but also *how* it should be done."

Another problem with the military, according to Dean, is that "the Commission frequently receives a Joint Chiefs' 'requirement' for a particular weapon or family of weapons, or a new reactor to produce a certain material, or a new production site, without any prior consultation between the Chiefs and the Commission. This is wrong. Proper liaison . . . is lacking. It is really not furnished by the Military Liaison Committee . . . What is needed today is . . . an unplugging of the

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channels of communication between the two agencies."

In several places Mr. Dean emphasized the importance to the U. S. of getting uranium from foreign sources. "In countries where there is no uranium program," he said, "the United States must be prepared, when new deposits are found, to make available . . . equipment. It must be prepared to lend money and furnish other inducements." In addition, "if we are to continue to get ore from abroad we must be prepared also to give up certain information . . . and technology which will aid these countries in . . . their own atomic programs." Actually, according to Mr. Dean, we could supply all the needs of our atomic energy program from our low-grade uranium sources, such as phosphate rock. "But," he says, "this uranium, because of the low grade of the ore, would cost several times as much as the uranium in Canada, the Congo, the Colorado plateau, and South Africa."

One very interesting chapter is on how a Commission expansion takes place. The steps described are: determination of the need, proving the need, securing Congressional approval, decision on whether to build a new site, decision on whether to build a new Government town, and selection of a site.

ALSO OF NOTE

Britain's Atomic Factories. This 100-page booklet presents the first picture of Britain's facilities for production of fissionable materials. The administrative set-up, with headquarters at Risley, and the operations of Springfields (ore processing), Windscale (plutonium production), and Capenhurst (gaseous diffusion plant) are described. *British Information Services, 30 Rockefeller Plaza, New York 20, N. Y., $1.25.*

Translation service. The Special Libraries Association, in cooperation with the John Crerar Library, has created a translation service based on a nucleus of 1,300 translations. Contributions may be made to the pool by deposition of good copies or 35-mm microfilm. Photoprint or microfilm copies of the translations listed may be obtained at the regular rates charged by the Crerar Library. A price list will be sent on request. *S. L. A. Translations Pool, John Crerar Library, 86 E. Randolph St., Chicago 1, Ill.*

NUCLEAR ENGINEERING

Penn State to Build Reactor

By W. M. BREAZEALE
The Pennsylvania State College
State College, Pennsylvania

CLEARING THE WAY for immediate construction, the trustees of the Pennsylvania State College have approved use of school funds for a research reactor. First operation of the reactor, with fuel allocated by the Atomic Energy Commission, is scheduled for early 1955.

Although AEC has approved the Penn State project, actual allocation is subject to the school's agreeing to observe AEC health, safety, and security rules.

The facility, to be built on the campus, will be modeled after the Bulk Shielding Facility at Oak Ridge National Laboratory (*1*). Estimated cost of the building, reactor, pool, utilities and roads is in the neighborhood of $300,000, exclusive of experimental equipment. The reactor will be housed in a steel frame bay 60-ft long, 30-ft wide, and 30-ft high. The framework will be covered with insulated metal siding. A two-story shop and office wing will be attached to the bay.

Initial operation will be at a power level of 100 kw. The facility is expected to be operated on an unclassified basis.

Unlike the Bulk Shielding Facility, as previously described (*1*), the reactor will be suspended from a movable bridge on which the control equipment will be located. The bridge will be mounted on wheels that rest on the parapet of the pool. The pool is 14-ft wide, 30-ft long, and 24-ft deep. Ample shielding is provided by 19 ft of water above the active lattice. A number of beam holes will extend through the pool walls. A movable gate within the pool will permit the reactor to be rolled to the far end of the pool, the gate closed, and the larger section of the pool drained. Personnel can then go into the drained portion to erect large pieces of equipment. Making the reactor movable also means that no shutters are required in the beam holes because the reactor can be moved sufficiently in the water to shield the beam holes from fission-product gamma rays after shutdown.

The facility will be built in the side of a hill. Only the beam-hole-end will be exposed. Earth will form the shielding on three sides of the pool and heavy concrete blocks will shield the exposed end. The building will extend beyond the beam holes to provide laboratory space around them.

The safety and control equipment will be similar to that described by Cole (*2*). Together, the three boron-filled control rods will be worth about 8% in reactivity. Period and flux-overload protection will be provided. The electronic safety circuits will be simplified by combining the preamplifier, sigma amplifier, magnet amplifier and period amplifier on one chassis. The regulating rod will be driven by a simple servo system much like that suggested by Cole.

The building and pool, to be built by a contractor, were designed by the university's architectural engineering department. The reactor itself, with the exception of the electronic control equipment and the fuel elements, will be fabricated in the university's shops. The facility will be located on the edge of the campus where room is available for the necessary exclusion area, a square about 250 ft on a side. The primary purpose of the fence is to exclude personnel who have not had indoctrination in health physics and reactor operation procedures.

Program for Reactor

Students in reactor engineering and reactor physics courses will receive training in reactor operation.

It is expected that a number of other activities will be inaugurated. These might include—analysis by radioactivation, radioisotope production, radiation damage studies, tracer work, medical applications, fission product utilization, neutron diffraction, etc.

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LETTERS

D_2O: Cost Is Key

DEAR SIR:

I have just received the December, 1953, NUCLEONICS and am rather disturbed to see on page 10 ["International Developments in Heavy Water Reactor Technology," a report of a meeting in Norway] a paragraph beginning:

"A less optimistic forecast for heavy water power reactors was made by John V. Dunworth of England."

This paragraph has been taken out of its context and leads to the conclusion that I do not see a future for heavy water reactors except for use in ship propulsion. I believe that this statement is true only when we are faced with purchasing heavy water at prices of the order of $100 per pound. However, there is every reason to believe that it is possible to make heavy water much more cheaply, and in this case I believe there may well be a future for land-based heavy water moderated power reactors for the generation of electricity.

—JOHN V. DUNWORTH
Atomic Energy Research Establishment
Harwell, England

NUCLEONICS omitted the words "unless its price were very drastically lowered" from the report it received, not realizing that Dr. Dunworth felt so optimistic about a drop in the price of heavy water.

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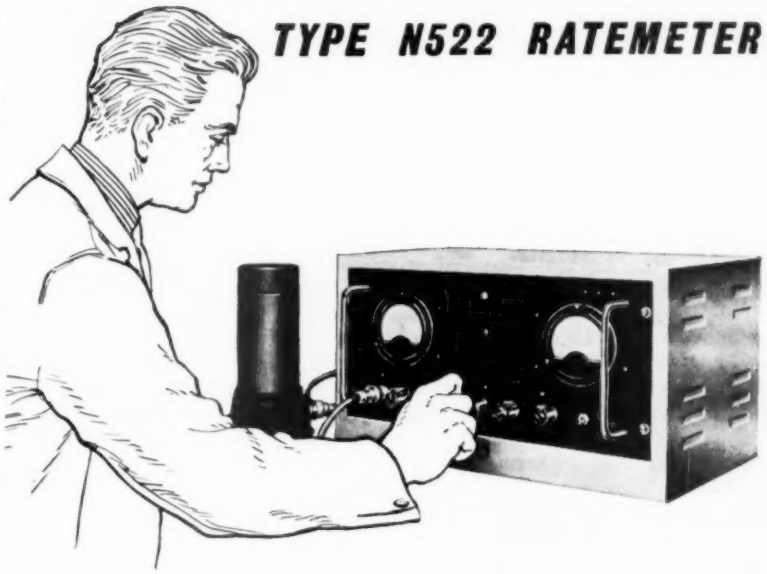

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NEWS

Eisenhower Submits Largest Atomic Energy Budget, Calls for End of Government Monopoly

A big boost to industrial development of nuclear power came in the President's first full-length budget message to Congress. In his official endorsement of legislation that would permit broader industry participation in such development, the President said the time has come to "depart from the federal government's existing monopoly in the field."

"Such widespread participation," he continued, "would be a stimulating and leavening force in this important field and will be consistent with the best traditions of American industrial development."

The proposed budget devotes $2,425-million to atomic energy, topping all previous yearly proposals.

The Presidential message stressed efforts to promote the peacetime use of atomic energy both in this country and in friendly foreign nations. "Reactor development," he said, "will be focused particularly upon the development of industrial atomic power for peacetime use."

The Atomic Energy Commission budget, drawn months before the message was prepared, did not appear to live up to this enthusiastic viewpoint.

Of the proposed $43.6-million expenditure for reactor development during the year beginning July 1, more than $35-million is earmarked for the utility-size pressurized water reactor already under development by Westinghouse Electric Corp.

Another $22.5 million was requested for "new research machines and facilities which are necessary to continued advances in atomic science."

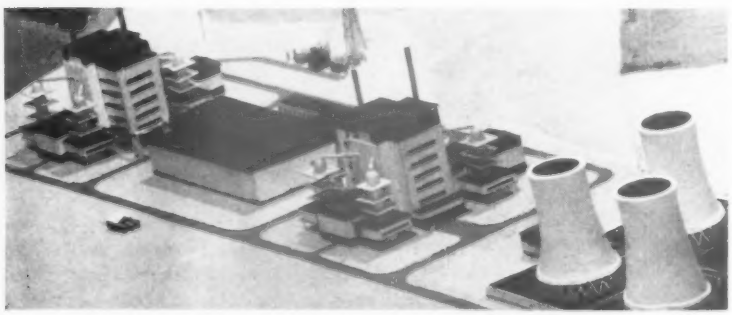
Research and development, including construction of experimental facilities, is to continue on several types of reactors that show promise of ultimately producing power at economic rates. However, the total funds that would be available for these purposes would be limited.

There was evidence in the budget message that nations supplying source materials are putting increasing pressure on the U. S. to share information on power reactor technology. President Eisenhower promised to submit to Congress proposals for relaxing present law to permit the exchange of both information and fissionable materials for nuclear power development.

AEC officials have long expressed the belief that freer exhange of information with such nations as Great Britain and Canada would contribute significantly to U. S. nuclear power activities.

This proposed exchange would be independent of the President's plan for an international agency to promote peaceful uses of atomic energy as outlined in the United Nations speech. Authority to fully participate in such an agency would be requested in separate legislation.

Under the budget submitted to Congress last month, the operating expenditures of AEC would rise from $912-million during the current fiscal year to $1.2-billion during the 1955 fiscal year. This is due primarily to expanded operations of the Oak Ridge, Paducah, Portsmouth and Hanford production facilities. Raw materials needed for these plants are now said to be available.



BRITISH POWER STATION model shows how reactor buildings flank turbine house. Fans in low wings on reactor buildings move heated CO_2 from reactors to vertical boilers outside. Steam from the boilers drives four turbo-generators in turbine house. Towers at right cool turbine condensate. The plant, being built at Calder Hall, will be finished about mid-'55

AEC Approves TVA Nuclear Power Study

The Tennessee Valley Authority, with the approval of the Atomic Energy Commission, will make a study of economic and technical aspects of producing nuclear power on a commercial basis.

The study will be similar to those being conducted by private industrial groups. Cost of the one-year study will be borne by TVA. Reports and recommendations will be submitted to AEC.

TVA supplies a major portion of power for AEC facilities at Oak Ridge and Paducah.

Midwestern Schools Plan New Cosmotron

A group of seven Midwestern educational institutions plan to cooperate on research projects using a proposed multi-million-volt cosmotron. The universities of Illinois, Minnesota, Wisconsin, Michigan, Indiana and Iowa and Iowa State College hope that the cosmotron and the projects will be financed by AEC or private foundations.

GE Preparing New Radiation Laboratory

A new general purpose radiation laboratory is being prepared at Schenectady by the General Electric Company. The laboratory will consolidate the company's expanding studies of industrial radiation techniques.

The facility, a part of GE's General Engineering Laboratory, will occupy a former factory building near the main plant. Conversion at a cost of $300,000 is under way.

The building will house six separate hot laboratories equipped for safe handling of radioactive material. Remote handling devices, monitoring equipment, and special clothing will be provided for personnel protection.

PHS Offers C^{14} Compounds for Medical Studies

A limited supply of radioactive corticosterone will be supplied to qualified investigators through funds from the U. S. Public Health Service's National Institute of Arthritis and Metabolic Diseases.

Other compounds, including cortisone acetate and intermediates obtained in its synthesis, are also available.

Requests in letter form describing the proposed research may be sent to the Endocrinology Study Section, Division of Research Grants, National Institutes of Health, Bethesda 14, Maryland. Transfer and use of the material are subject to Atomic Energy Commission regulations.

Brookhaven to Build 25-Bev Synchrotron

The Atomic Energy Commission has approved design and construction of a 25-Bev alternating gradient synchrotron at Brookhaven National Laboratory, Upton, N. Y. It will be completed in five or six years.

The $20-million synchrotron will use a series of alternate strongly converging and diverging magnetic fields to confine the particle beam in a tube of relatively small cross section. This allows the production of high-energy beams with smaller electromagnets and related equipment than would otherwise be possible (NU, Nov. '52, p. 106).

The focusing is done by magnetic fields as in present racetrack-design cyclotrons. But instead of having the field gradient in the same direction in all sections of the magnet, the direction of the field gradient alternates from section to section—the field increases towards the center of the cyclotron in one magnet section and increases towards the outside in the next section.

This alternation of converging and diverging forces is found to have a cu-

The Editors Hear

There is speculation that the President's international atomic pool proposal may sidetrack domestic atomic energy legislation. At this writing this does not seem to be the case. A handful of bills (on industrial participation, patents, technical information, organizational details, etc.) have gone through the hands of the commissioners of AEC and are going the rounds of various governmental agencies. They should reach the Congress by early February, with hearings on industrial power legislation to start by mid or late February.

●

Some things that affect the domestic atomic energy scene may get wrapped up in international diplomacy. For example, a report on the U. S. reactor development program was promised by the AEC in 1949. A number of attempts have been made to produce such a report. The latest (last summer for the Joint Committee power hearings), although it was said to contain no classified information, was pulled back at the last minute for policy reasons. Now it appears that what some call an innocuous report may be held for release at some strategic time in negotiations between East and West.

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With the rapidly growing university interest in research reactors (about 20 schools are at various stages of planning), legislative action is necessary to legalize university possession of more than bomb quantities of enriched fissionable material. In the interim, AEC must provide a government custodian on the campus.

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A testimonial to the positive interest of industry in atomic energy will be the extensive activities of one nuclear power group. It is reported that the Dow-Detroit Edison team of nearly 30 companies will spend nearly $2-million in 1954—this before any change in the Atomic Energy Act.

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The AEC Senior Responsible Reviewers were scheduled to meet the last week in January to review declassification policy from top to bottom. This was in anticipation of another tripartite (with Britain and Canada) declassification conference later this year . . . . The six-man U. S. group is one short with R. H. Crist of Carbide and Carbon having resigned recently. Another member, J. R. Richardson, of the University of California, is on leave with the London office of ONR.

●

The midwestern group of universities that combined to build a large accelerator may not get the AEC money it is seeking. If AEC builds a large machine in the Midwest, it will probably be on the Argonne National Laboratory reservation rather than on the university-selected site. The schools are trying to raise their own money ($20 million) . . . . For the same reason (apparently to assure continued interest in the national labs) AEC decided to locate the recently announced 25-Bev machine at Brookhaven instead of on the campus of one of two universities that wanted it.

●

A program for the development of one or more "package" nuclear power plants suitable for use at military installations is about ready for presentation to the five-man AEC. If approved, there may be a drive to get a supplementary budget request through in time for the fiscal year starting July 1. The first such plant would undoubtedly be built stateside for convenience, training of personnel, etc. If the job is put out on bid to industry, there are four companies that may bid on a lump-sum basis. It is expected that the capacity of the plant would be no bigger than 20,000 electrical kw.

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The AEC radiation instruments branch is carrying out a survey to see what the comparison is between AEC buying of commercial radiation instruments and AEC building its own.

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The so-called AEC five-year program of reactor development, prepared at the request of the Joint Committee on Atomic Energy, has not yet been approved by the Commission. The program calls for four reactors: a 10,000 kw (heat) sodium reactor experiment (SRE); a 1,000 kw (elec.) boiling water reactor; a 50 megawatt (heat) breeder; and a 50 megawatt (heat) homogeneous unit. A fifth item, which some call the key to economic nuclear power, is a fabrication plant for plutonium fuel elements.

mulative convergent effect, analogous to the convergence of light by alternating converging and diverging lenses. The convergence of the beam of particles is so strong that the vacuum chamber in which the particles travel can be small compared to the size of chamber necessary in large cyclotrons today.

The small vacuum chamber permits the magnet around it to be quite small. The magnet units of the cosmotron now operating at Brookhaven are 8×8 ft in cross section; magnet sections for the synchrotron will be only 2×3 ft in cross section. This will produce economies in the design.

The orbit diameter of the new 25-Bev accelerator will be 600 ft as compared to the 60-ft diameter of the 3-Bev cosmotron.

RAW MATERIALS

• **Australia.** Surface indications suggest an extremely rich uranium find in the Norseman area of Western Australia.

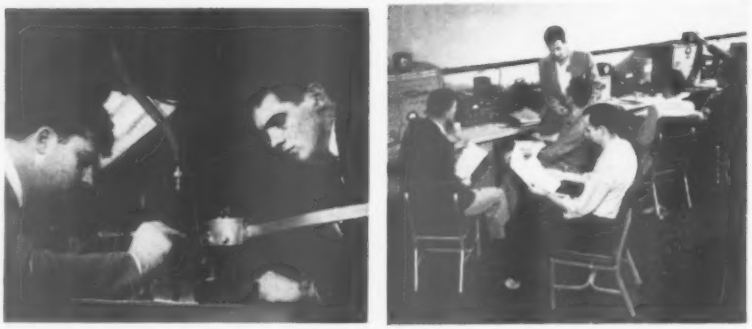
• **New Mexico.** A $3-million uranium-ore-treating plant will be built at Shiprock, N. M., by Western-Knapp Engineering Division of Western Machinery Co. The plant, first to use the acidulation process for separating uranium oxide from carnotite, will also be designed and equipped by the firm. It is expected to be completed by December, 1954.

• **Central Africa.** Uranium deposits that had been originally registered in 1951 have been taken over by a South African mining group. If a thorough investigation now being made of a 50-square-mile area provides satisfactory results, the first mine in Central Africa to produce only radioactive materials will be established. Recovery plant is estimated to cost not less than $3-million.

• **South Africa.** The country's fourth and largest uranium plant has recently been completed. Including a sulphuric acid unit, the plant, at Western Reefs Mine near Orkney, cost about $13.8-million.

Some of the plants in South Africa have been expanded and 23 gold mines have been added to the country's uranium program. Total capital investment in the program is now about $140-million. Annual gross revenue is expected to exceed $84-million.

Combining work . . . with classroom is . . .



the way reactor engineers are trained as . . .

ORSORT Prepares for 300th Student

With March 15 set as the deadline for applications, Oak Ridge School of Reactor Technology is planning to accept 80 students for its 1954–1955 session. Including the group now in training, ORSORT has trained over 200 reactor engineers since its inception in 1950.

The curriculum includes courses in reactor analysis, reactor engineering, reactor technology, experimental reactor physics, reactor materials, reactor chemistry, reactor nuclear physics, and reactor design. Students also participate in actual reactor research, design and construction at Oak Ridge National Laboratory. Both ORSORT and ORNL are operated for the Atomic Energy Commission by Carbide and Carbon Chemicals Co.

Selection of students is on a competitive basis. Industrial companies can sponsor members of their technical staffs as ORSORT students. College graduates in the fields of chemistry, physics, engineering, engineering physics, and metallurgy—who are not employed in industry—receive a monthly salary while attending ORSORT. No tuition is charged the students.

Additional information and application forms are available from ORSORT, Post Office Box "P," Oak Ridge, Tenn.

NEWSMAKERS

Logan B. Emlet, formerly superintendent of the Y-12 electromagnetic separation plant at Oak Ridge, has been appointed manager of production for the Y-12 plant and the gaseous diffusion plants at Oak Ridge and Paducah. He succeeds **W. B. Humes. J. P. Murray** becomes superintendent of the Y-12 plant. **R. G. Jordan,** formerly assistant superintendent at Paducah, succeeds Murray as superintendent of the gaseous diffusion plant there.

Herbert Insley has retired from his position as chief of the mineral products division of the National Bureau of Standards. A pioneer in the application of petrographic microscopy to ceramic problems, he will do consulting.

Lieut. Gen. Sir Frederick Morgan, controller of atomic energy in the British Ministry of Supply since 1951, has been appointed controller of atomic weapons.

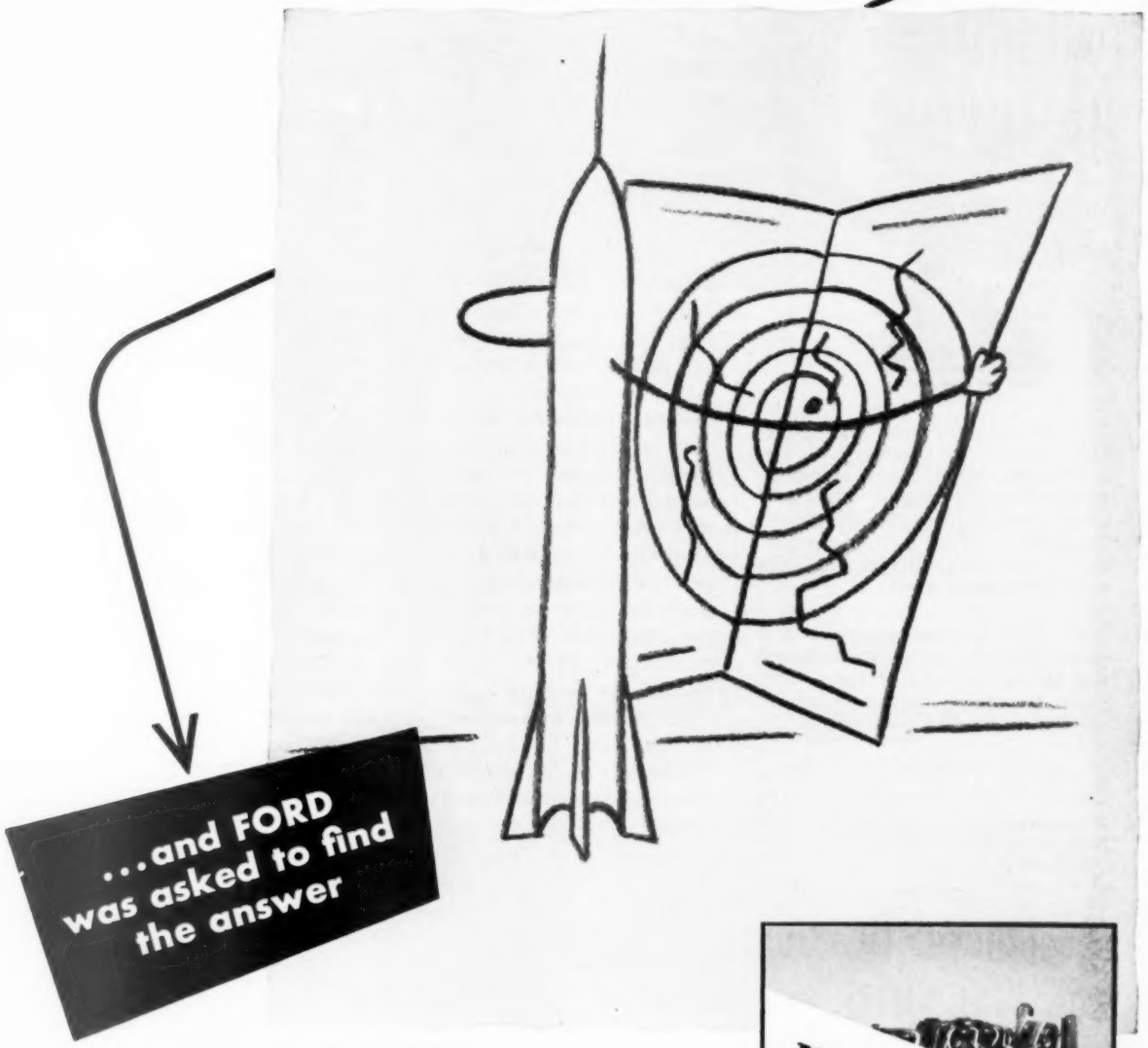
Ralph R. Sachs, chief of the public health and welfare section of the Hanford Works, has recently been chosen vice-president of the Western Branch of the American Public Health Association.

William L. Davidson has resigned as director of industrial development for AEC to become associated with Food Machinery and Chemical Corp. **David C. Beckler,** assistant director, has also resigned and will become scientific advisor in the Office of Defense Mobilization.

C. E. Stevenson, formerly associate director of the chemical engineering division of Argonne National Laboratory, has been named technical director of the chemical processing plant at the Phillips Petroleum Co's Arco plant.

I. F. Richardson, Jr. has been appointed assistant general manager of the Kansas City Division of Bendix Aviation Corporation. The division is the center of Bendix activities in the

HOW TO TEACH A MISSILE
to read a map



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News Sidelights

Physical research contracts. Forty-two unclassified physical research contracts with universities, private research institutions, and industrial laboratories have been announced by the U. S. Atomic Energy Commission. Research includes accelerator and cyclotron work and investigations of behavior of materials at extreme temperatures, generation of high voltages by nuclear radiations, gaseous chemical reaction kinetics, metal-ceramic bonding, and uranium accumulation in vegetation.

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Tumor-treatment reactor proposed. Based on initial experimental work by the Brookhaven National Laboratory medical department, the laboratory has made an informal proposal to AEC for a small reactor for treatment of malignant tumors.

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Pakistan laboratory. The Pakistan government will accept the British government's offer to provide, under the Colombo Plan, free equipment for a nuclear laboratory for the Pakistan Council of Scientific and Industrial Research.

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Progress in India. A 1.2-Mev Cockcroft-Walton generator is operating in temporary quarters of the Tata Institute for Fundamental Research at its new home near Colaba. It will be used for fundamental research and to obtain data for reactor design. In the belief that science does not flourish in an atmosphere of secrecy, Prime Minister Nehru stated that there would be no secrecy about the peaceful atomic energy work of the institute. He made the statement during ceremonies connected with the laying of the foundation stone for the institute's permanent home.

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Award for isotope application. An annual award of $1,000 for "outstanding contribution to nuclear isotopic applications in the field of chemistry" has been established by the Nuclear Instrument and Chemical Corporation of Chicago. It will be administered by the American Chemical Society.

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Water-boiler reactor delivered. North American Aviation's atomic energy research organization has recently delivered a 100-watt water-boiler-type research reactor to the California Research Company's Livermore labs. The unit, second of this type built by NAA, is fueled by about four gallons of an aqueous uranyl-sulfate solution within a one-ft-diameter sphere.

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Lithium in thermonuclear weapons? In estimating that the cost of tritium produced in the $1.5-billion Savannah River plant "will not be much under $1-million a pound," Ralph E. Lapp, in a recent newspaper article, postulated that a form of lithium or some other light metal might be the main source of energy in the thermonuclear reaction. The tritium-deuterium mixture, itself "triggered" by heat from a conventional atomic bomb, would be only a trigger for the lithium reaction.

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Italian synchrotrons. A 100-Mev unit is now being built in Turin; a 1-Bev machine is being planned for construction near Milan.

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Another university reactor approved. The Atomic Energy Commission has agreed to lend the Pennsylvania State College uranium for its already-designed research reactor. It will probably begin operation in early '55. (For details, see p. 72)

nuclear energy field. Richardson was formerly manager of aircraft equipment sales and service for the corporation's Bendix Products Division, South Bend, Ind.

George E. Uhlenbeck of the University of Michigan and **Samuel A. Goudsmit** of Brookhaven National Laboratory have been honored by a Research Corporation award for their discovery of electron spin 25 years ago.

Ernst Henry Krause, superintendent of the radiation division of the Naval Research Laboratory, has been named associate director of research. He succeeds **F. N. D. Kurie,** who has been selected as superintending scientist of the Naval Electronics Laboratory in San Diego, California.

Serge A. Korff of New York University has been appointed reporter for cosmic rays on the National Research Council committee responsible for American contributions to a year-long program of concentrated international investigations. The studies will include meteorology, oceanography, terrestial physics, astronomy, and other fields in which international cooperation is essential.

MEETINGS

February 22–25: Industrial Ventilation Conference; Michigan State College, East Lansing, Mich. Talks and demonstrations on the practical aspects of ventilation engineering, including the equipment used for measuring and evaluating airflow rates, hood design, dust collection, and system design. Both general and specific problems will be considered.

March 24: Institute of Radio Engineers National Convention (NU, Jan. '54, p. 74).

May 17–19: Radiation Research Society Annual Meeting; Wade Park Manor Hotel and Western Reserve University, Cleveland Ohio. A symposium on the role of nuclear damage in radiation induced cell injury and death will be held. Invited and submitted papers will be presented on radiation physics and chemistry. Information may be obtained from Dr. A. Edelmann, Brookhaven National Laboratory, Upton, Long Island, N. Y.

June 20–23: American Institute of Chemical Engineers' Conference on Nuclear Engineering (NU, Aug. '53, p. 64).

July 13–17: Harwell Conference on Experimental and Theoretical Nuclear Physics (NU, Jan. '54, p. 77)

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MODEL	RADIATION SPECTRUM (Kev. effective)	SENSITIVITY RANGES	TOTAL DOSE (roentgens)	USE
601	30 to 400	10-30-100 r/min.	1 to 999	X-ray therapy
602	30 to 400	100-300-1000 r/min.	10-9990	X-ray therapy
603	400 to 2000	5000-15000-50000 r/min.	500-499, 500	On telecurie Gamma sources
604	20 to 50	100-300-1000 r/min.	10-9990	Low energy therapy
605	400 to 2000	1-3-10 r/min.	0.1-99.9	Gamma ray sources
606	400 to 2000	10-30-100 r/min	1-999	Gamma ray sources
607	400 to 2000	100-300-1000 r/min.	10-9990	Gamma ray sources
608	30 to 400	10-30-100 r/hr.	1-999	Survey in vicinity of X-ray machines
609	**Not Applicable**	10-30-100 r/min.	10-9990	Beta-Gamma skin dose
610	I^{131} and AU^{198}	1-3-10 mc. I^{131}	**Not Applicable**	Isotope assay
611	400-2000	0.5-1.5-5 r/min.	0.05-49.95	Gamma ray sources

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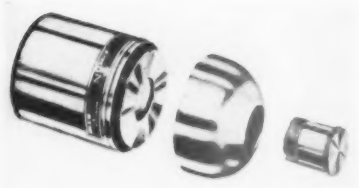
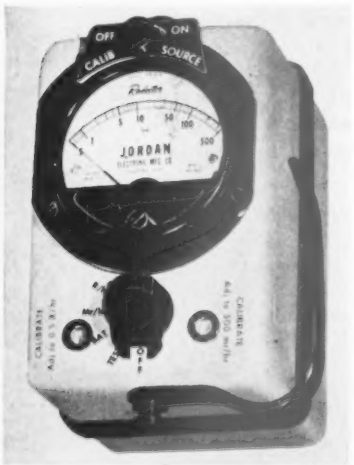


Scintillation Counter Head

Nuclear Research and Development, Inc., 6425 Etzel Ave., St. Louis 14, Mo. The SC-44 combination wide angle and collimated medical head is an attachment for the SC-3 scintillation counter. Quick conversion from a wide-angle counter to a highly collimated counter is provided by the round-nose section, which is attached to the counter by a bayonet fitting. The nose section is threaded to accept any additional collimators up to 1 in. bore. Two sizes of NaI crystals are available, both 1 in. long with $\frac{3}{4}$-in. and 1-in. diameters. (P-1)

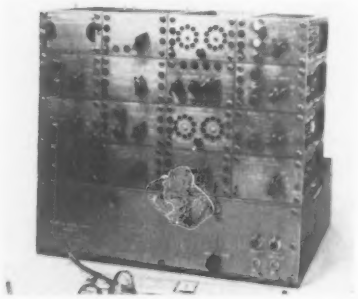


Beta-Gamma Monitor

Jordan Electronic Sales, Inc., 119 E. Union St., Pasadena 1, Calif. Two Radector models are available: the AGB-50-SR has ranges of 0.05–50 mr/hr and 0.05–50 r/hr; the AGB-500-SR has ranges of 0.5–500 mr/hr and 0.5–500 r/hr. NBS tests are said to have shown it to be substantially γ-ray energy independent between 75 kev and 1.3 Mev, directionally independent, and with a battery life of 400 hr continuous use without change of calibration. The instrument, operated by one-knob control, is hermetically sealed, immersion proof, shock mounted, and unaffected by temperature, pressure, or humidity. (P-2)

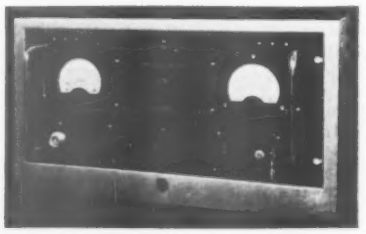
Laboratory Mixer

Mixing Equipment Co., Inc., 135 Mt. Read Blvd., Rochester 11, N. Y. The Lightnin laboratory mixer is designed for fluid mixing, blending, solids suspension, and gas dispersion. Low-viscosity fluids can be handled in quantities up to 50 gal. It clamps onto the rim of any vessel and adjusts to any angle. The $\frac{1}{20}$-hp motor turns the shaft at 1,750 rpm. Total weight is 22 lb. (P-3)



Electronic Pulse System

Audio Products Corp., 2101 W. Olive Ave., Los Angeles, Calif. The Modular System consists of 16 digital pulse units for performing the basic functions of digital-pulse operations. A complete system has capability of 72 separate functions, with as many as 31 functions simultaneously available. Each unit is $2\frac{3}{4} \times 4\frac{1}{2} \times 9$ in. (P-4)

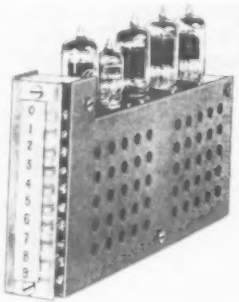


Counting-Rate Meter

Ekco Electronics Ltd., Southend-on-Sea, Essex, England. The type N522 counting-rate meter incorporates an input amplifier and built-in variable high-voltage supply for operating a G-M tube preamplifier or scintillation counter amplifier. Counting rate is 0–10,000 pulses/sec. Time constant and deadtime are variable. (P-5)

Fluorescent Chemicals

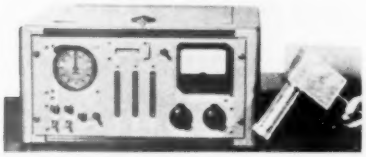
Pilot Chemicals, Inc., 47 Felton St., Waltham 54, Mass. Fluors added to the company's line of fluorescent chemicals are: diphenyl oxazole, naphthyl phenyl oxazole, diphenyl acetylene, and stilbene. (P-6)



Electronic Counter

Tracerlab, Inc., 130 High St., Boston 10, Mass. The SC-41 plug-in, direct-reading counter is capable of speeds up to 75,000 cps with a 5-μsec resolving time. The counter has a 5-tube decade scaling unit with a self-contained 10-light neon indicator. Four conventional scales-of-two with additional circuit connections cause the system to deliver an output pulse and to reset after 10 input counts have been received. Input pulse range is 75–100 volts, negative, with 1-μsec risetime or less. Output is 100 volts, negative, at

count of 10, with 100-volt positive pulses available at counts of 2, 4, 6, and 10 for special uses. Power requirements are 250–400 volts, d-c, with 300 volts at 14 ma nominal, and 6.3 volts, a-c, at 1.5 amperes. Connector is a Jones P-310-AB. (P-7)

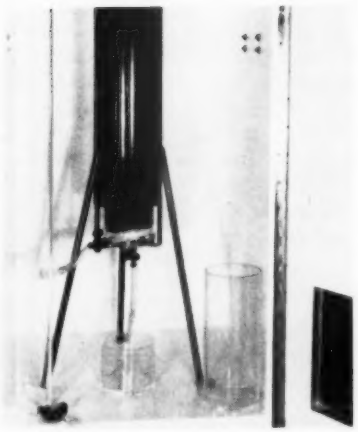


Counting System

Radiation Instrument Development Laboratory, 2337 W. 67th St., Chicago 36, Ill. The model 15 counting system, equipped with a Veeder Root register, and Standard Electric timer, accommodates anthracene, sodium iodide, and other crystals, including thermal-neutron crystals. Direct-coupled amplifier has gain of 2,500 and overloads without overshoots; stability factor is greater than 100. Amplifier output feeds a discriminator so that the unit may be used as an integral-type pulse-height analyzer. Discriminator control is a ten-turn Helipot. High voltage is 500–1,500 volts, negative; regulation is better than 0.005 for 1% change in line voltage. (P-8)

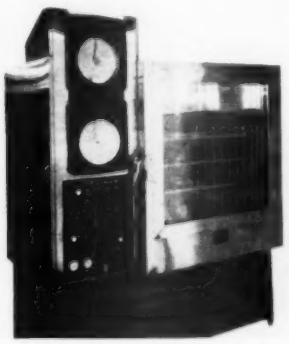
Oscilloscope Timer

American Electronic Laboratories, Inc., 641 Arch St., Philadelphia, Pa. The Raster Timer enables times measurements of recurrent or transient phenoma by providing a series of 10 calibrated sweeps, one above the other on the screen; assuming a useful area of 4×4 in., the equivalent of a 40-in. viewing screen is created on the horizontal axis. Sweeps provided by the timer are 1, 3, 5, or 10 msec each, crystal-controlled to 0.01 msec. Each of the 10 sweeps carries time markers allowing measurements to 0.1–0.05% of total presented sweep time. (P-9)



Radioactive-Liquid Filter

Atomic Energy Waste Disposal Service, Box 255, Oakland 4, Calif. The Nuclear Liquid Filter has built-in shielding and effluent hold-up tank and an indicator section to show whether the effluent is acidic or basic. In the case of strongly ionized radioactive salt, e.g., cesium tracer, the cartridge decontaminates many thousands of gallons of liquid wastes; with inactive salts equivalent to 100 ppm sodium nitrate, it decontaminates many hundreds of gallons. However, if 1,000 ppm of inactive salts are present, the volume decontaminated is of the order of 50–75 gal. Wastes that have been treated include Cs^{137}, Sr^{90}, Co^{60}, Ni^{63}, Ru^{106}, Ce^{144}, S^{35}, C^{14}, and fission products. If shipping charges are prepaid, Disposal Service will dispose of spent cartridges. (P-10)



Atmosphere-Control Cabinet

Package Materials Laboratories, Inc., Dept. FM-3, 461 Crescent Rd., Hatboro, Pa. The 15-CF atmosphere control cabinet is designed for moisture-vapor testing within the limits of −20° F to 200° F, with a relative humidity of 18% to slightly under 100%. The unit may be ordered equipped with conventional or electronic controls. (P-11)

Integrating Ratemeter

Victoreen Instrument Co., 5806 Hough Ave., Cleveland 3, Ohio. The model 575 Radocon measures in roentgens both rate and total dose. The standard probe covers 30–400 kev, 1–999 r total dose in 1-r steps, and intensities of 1–100 r/min in ranges of 0–10, 0–30, and 0–100. Other chambers cover energies up to 2 Mev; probes are available for total dose and intensities 1,000 times higher and lower than those of the standard probe. A cable up to 200 ft long connects the probe to the control unit. (P-12)

Radiation Monitor

Measurement Engineering Ltd., Arnprior, Ontario, Canada. This airborne radiation survey instrument was designed by the Chalk River Laboratories and Flight Research Section of the National Research Council of Canada. Two matched detector units are comprised of RCA 5819 photo multipliers light-coupled through sili-

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This is the only book in English since the war that covers the field of nuclear moments for the stable nuclei and for the radio-active nuclei whose lifetimes are sufficiently long to be measured by techniques similar to those used with stable nuclei. Based on the author's section in Volume I of *Experimental Nuclear Physics*, edited by Emilio Segrè, the work gives a detailed discussion of both the experimental and theoretical aspects of the measurement techniques and of the observed results . . . examining nuclear measurements as they affect the fields of nuclear physics, chemistry and solid state physics.

Dec., 1953, 169 pages, $5.00

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WILLIAM V. SMITH,
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and RALPH TRAMBARULO,
University of Delaware

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1953, 446 pages, illus., $8.00

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Vol. I, 1953, 789 pages, 275 illus., $15.00
Vol. II, 1953, 600 pages, 152 illus., $12.00

THEORETICAL NUCLEAR PHYSICS

by John M. Blatt, *Univ. of Sydney* and Victor Weisskopf, *M.I.T.*

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cone oil to NaI(Tl) crystals 2 in. in diameter and 4 in. long. A filter network provides high resolution by removing rapid fluctuations from the photomultiplier outputs. Optimum survey height is 550 ft; field of view is limited by distance to about 30 deg. Cosmic radiation is cancelled by a discrimination circuit. Counting ranges are 0–200 to 0–20,000 cpm; accuracy is ±5%. (P-13)

Darkroom Insert Tank

Bar-Ray Products, Inc., 209 25th St., Brooklyn 32, N. Y. These insert tanks can be hung on any Bar-Ray master tank by means of a special stainless-steel lip welded to the insert. The tank may be used for wetting agent, as an empty drip compartment for hangers before being put into the dryer, or for additional developer or fixer space. They are constructed of heavy-gage type 316 stainless steel. The two available sizes are $20\frac{1}{2}$ in. deep and $14\frac{3}{4}$ in. long; the B635 ($2\frac{1}{2}$-gal capacity) is $2\frac{1}{2}$ in. wide and the B636 (5-gal capacity) is $5\frac{3}{4}$ in. wide. (P-14)

Gamma-Ray Fluoroscope

Radioactive Products, Inc., 443 W. Congress, Detroit 26, Mich. This γ-ray fluoroscope, designed for industrial applications, utilizes three Co

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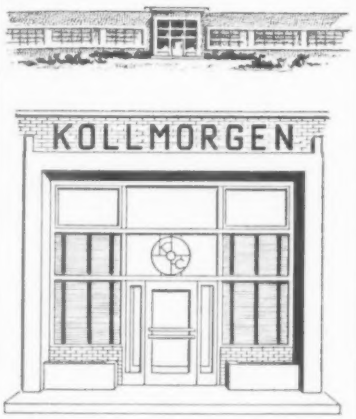

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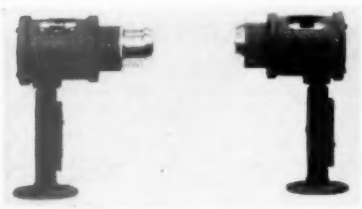
pellets 1 cm in diameter and 1 cm long. The unit is 7 ft high and weighs 4,500 lb. The source is contained in a steel-encased cylindrical lead shield 18 in. in diameter and 2 in. thick. The operator peers through a lead-glass window at a mirror tilted 45 deg; when the piece to be examined is in place on the work table, the operator pulls an exposure lever that retracts a shutter under the cobalt source, and a shadowgraph is visible to the operator. Radioactive Products leases out the fluoroscope and will service it and replace the source when its power is spent. (P-15)

Automatic Scaler

Beckman Instruments, Inc., Berkeley Scientific Div., 2200 Wright Ave., Richmond, Calif. The model 2200 automatic scaling unit consists of an input pulse-height discriminator, an electronic scaling channel with 1-μsec paired-pulse resolution, mechanical register, time clock, and automatic control circuitry; high voltage is not supplied. The scaling channel has 4 plug-in decimal counting units. Count indication is direct reading. Maximum counting speed is 100,000 sec. A 4-digit mechanical register is used and provision is made for driving an external relay in parallel with the register. The instrument provides predetermined time, predetermined count, and motor-driven reset. (P-16)



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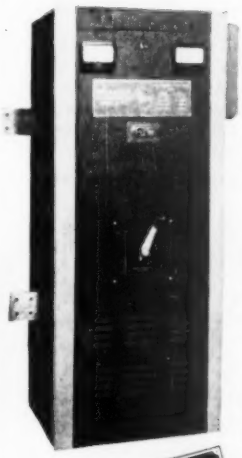

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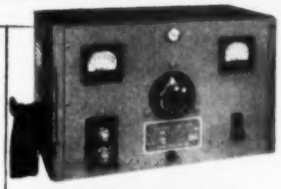

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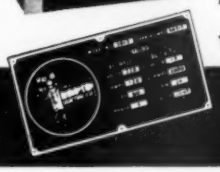

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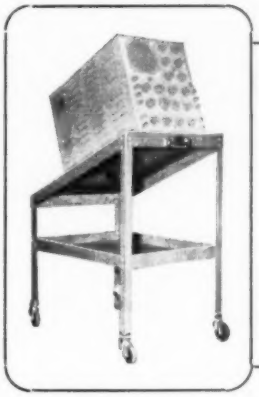

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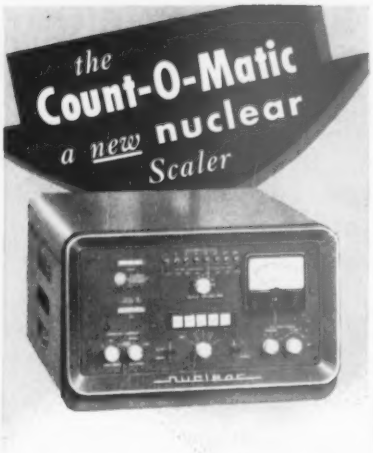

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Nuclear Instrument & Chemical Corporation, 235 W. Erie St., Chicago 10, Ill.

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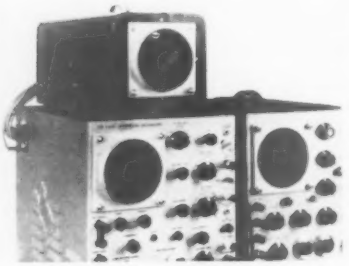
vernier focusing adjustment permit distant operation employing a parallel light beam or short pin-point operations. Lenses in both light source and photoelectric tube housing provide highly directional response. Lens position can be locked at any setting. Units are provided with a light-sensitivity control. (P-17)



Oscilloscope Converter

Computer Control Co., 106 Concord Ave., Belmont 78, Mass. The model 3CT1 dual-beam conversion unit is developed for use with ¾ Tektronix models 511, 512, and 514 oscilloscopes. After conversion, the oscilloscopes can be used for normal single-beam operation or with the converter. The twin-gun cathode-ray tube is the DuMont 58P1. Available phosphors are 1, 2, 7, and 11. (P-18)

Clinical Cart

Nuclear Instrument & Chemical Corp., 229 W. Erie St., Chicago 10, Ill. The model CA4 cart transports equipment for clinical radiation measurements. It is designed to carry a shielded radiation detector and scaler, rate meter, or recorder. The flexible probe arm extends to 44 in., carries up to 23 lb, and is counter-poised to keep the detector in position during uptake studies. A clamp at the end of the arm holds a 3-in.-diameter detector or a smaller de-

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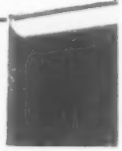

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INDUSTRY NOTES

▶**Blaw-Knox Co.,** Pittsburgh, Pa., has received an AEC order for 1,240 tons of prefabricated piping for the Portsmouth, Ohio, gaseous diffusion plant. Amounting to $1,159,061, this is the third piping order received by Blaw-Knox for this project. The three awards total over $3-million and cover 3,040 tons of special piping.

▶**Vitro Manufacturing Co.,** and its two subsidiaries, Vitro Corporation of America and Vitro Chemical Co., will merge into one corporation. The new corporation will be known as **Vitro Corporation of America** with offices at 233 Broadway, New York 7, N. Y. Five divisions of the corporation will be Vitro Manufacturing Co., Pittsburgh (ceramic colors and chemical products); Vitro Uranium Co., Salt Lake City (uranium ore processing); Canonsburg

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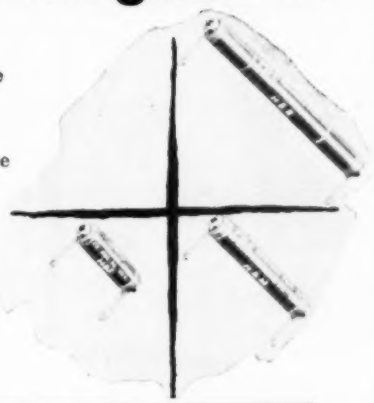

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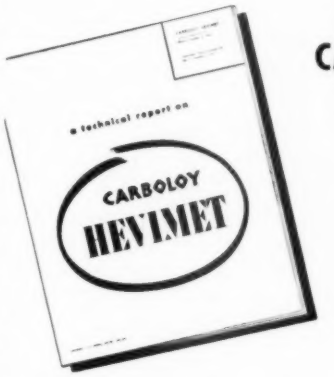

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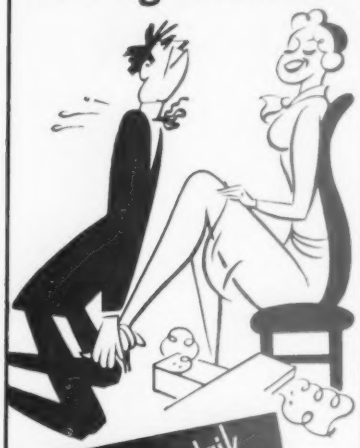

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Rare Metals Co., Canonsburg (metal refinery and recovery); Vitro Laboratories, Silver Spring (research and development); and Vitro Engineering Div., New York (engineering, design and construction management services).

▶**Tracerlab, Inc.,** 130 High St., Boston 10, Mass., has received a $1.5-million order from the U. S. Navy for scintillation-type radiation survey meters and spare parts.

▶**General Electric Co., X-Ray Department,** Milwaukee, Wis., will manufacture six-million-volt linear accelerators under a 10-year agreement with Stanford University. These accelerators will be a smaller version of the one-billion-volt accelerator now nearing completion at Stanford. They should be on the market within a few years.

▶**International Testing Service** is a newly formed division of Jackson and Church Co., Saginaw, Mich., to provide engineering services to firms facing problems in testing, instrumentation and related fields of research and development.

LITERATURE AVAILABLE

Automatic scaler. Specification sheet describes the model 183 scaler for use with G-M or scintillation counters. *Nuclear Instrument and Chemical Corp., 229 W. Erie St., Chicago 10, Ill.* (L-1)

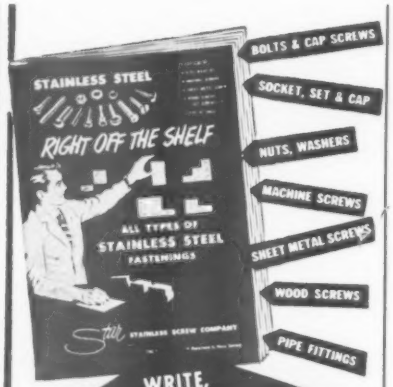
Roll film camera. Catalog EML418 describes electron microscope camera. *Engineering Products Dept., RCA Victor Div., Camden 2, N. J.* (L-2)

X-ray and analytical equipment. "Research and Control Instruments" covers X-ray diffraction, spectrometry and spectrography, and components and accessories such as tubes, rectifiers and cameras. *Research & Control Instruments Div., North American Philips Co., Inc., 750 S. Fulton Ave., Mount Vernon, N. Y.* (L-3)

Boron-carbon resistors. Bulletin B-8 gives data on characteristics, applications, tolerance, windings, terminations, dimensions, and insulation. *International Resistance Co., 401 N. Broad St., Philadelphia 8, Pa.* (L-4)

Plant stream analyzer. Sheet describes modification of model 21 infrared gas analyzer. *Liston-Becker Instrument Co., Inc., 649 Hope St., Stamford, Conn.* (L-5)

Laboratory equipment. Catalog SD-2 suggests laboratory layouts and lists



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accessories. *Metalab Equipment Corp., 214 Duffy Ave., Hicksville, N. Y.* (L-6)

Pumps and accessories. Catalog 13-K describes glass-blowing equipment, high-vacuum pumps, accessories. *Research Vacuum Supply Co., 3434 Montrose Ave., Chicago 18, Ill.* (L-7)

Cathode-follower probe. Model F-400 probe, for coupling high-impedance dynamic voltage generators into electronic measuring equipment, is described in Bulletin F-400. *Gulton Mfg. Corp., Metuchen, N. J.* (L-8)

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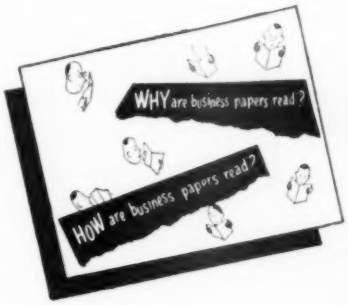

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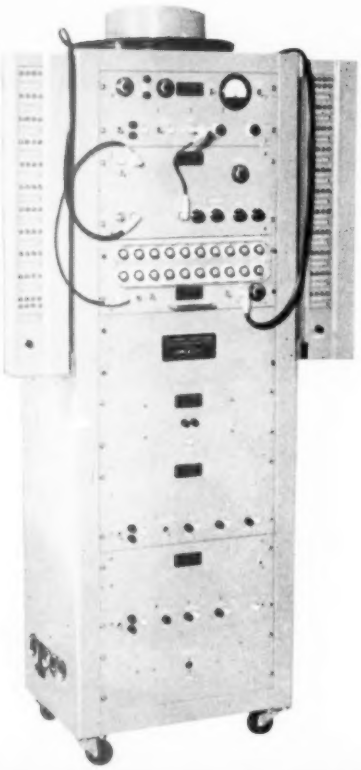

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MODEL 510

SPECIFICATIONS

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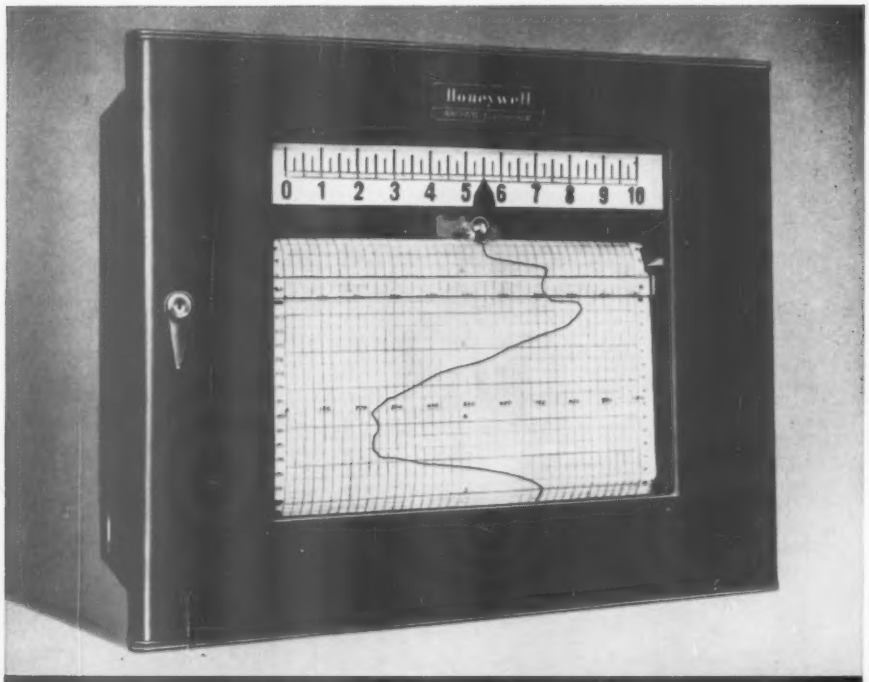

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